3. *Date:* September 26, 2006.
Time: 9 a.m. to 5 p.m.
Room: 415.
Program: This meeting will review applications for U.S. History, submitted to the Division of Preservation and Access at the July 25, 2006 deadline.

Heather Gottry,
Acting Advisory Committee Management Officer.

[FR Doc. E6–15021 Filed 9–11–06; 8:45 am]

BILLING CODE 7536–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting Notice

DATE: Weeks of September 11, 18, 25, October 2, 9, 16, 2006.

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

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of September 11, 2006

Monday, September 11, 2006

9:30 a.m.
Discussion of Security Issues (Closed—Ex. 1).
1:30 p.m.
Discussion of Security Issues (Closed—Ex. 1 & 3).

Tuesday, September 12, 2006

9:30 a.m.
Meeting with Organization of Agreement States (OAS) and Conference of Radiation Control Program Directors (CRCPD) (Public Meeting) (Contact: Shawn Smith, 301–415–2620).
This meeting will be webcast live at the Web address—*http://www.nrc.gov.*
1 p.m.
Discussion of Security Issues (Closed—Ex. 1).

Week of September 18, 2006—Tentative

There are no meetings scheduled for the Week of September 18, 2006.

Week of September 25, 2006—Tentative

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

Week of October 2, 2006—Tentative

There are no meetings scheduled for the Week of October 2, 2006.

Week of October 9, 2006—Tentative

There are no meetings scheduled for the Week of October 9, 2006.

Week of October 16, 2006—Tentative

Monday, October 16, 2006

9:30 a.m.
Briefing on Status of New Reactor Issues—Combined Operating Licenses (COLS) (morning session).

1:30 p.m.
Briefing on Status of New Reactor Issues—Combined Operating Licenses (COLS) (afternoon session).
(Public Meetings) (Contact: Dave Matthews, 301–415–1199).

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

Friday, October 20, 2006

2:30 p.m.
Meeting with Advisory Committee on Reactor Safeguards (ACRS) (Public Meeting) (Contact: John Larkins, 301–415–7360).

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

The schedule for Commission meeting 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: September 7, 2006.

R. Michelle Schroll,
Office of the Secretary.

[FR Doc. 06–7603 Filed 9–8–06; 9:57 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 August 18, 2006 to August 31, 2006. The last biweekly notice was published on August 29, 2006 (71 FR 51222).

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

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination. Within 60 days after the date of publication of this notice, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene.

Normally, the Commission will not issue the amendment until the expiration of 60 days after the date of publication of this notice. The Commission may issue the license amendment before expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. In addition, the Commission may issue the amendment prior to the expiration of the 30-day comment period should circumstances change during the 30-day comment period such that failure to act in a timely way would result, for example in derating or shutdown of the facility. Should the Commission take action prior to the expiration of either the comment period or the notice period, it will publish in the **Federal Register** a notice of issuance. Should the Commission make a final No Significant Hazards Consideration Determination, any hearing will take place after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

Within 60 days after the date of publication of this notice, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If a request for a hearing or petition for leave to intervene is filed within 60 days, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing.

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment. If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed by: (1) First class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff; (2) courier, express mail, and expedited delivery services: Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff; (3) E-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, *HearingDocket@nrc.gov*; or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC, Attention: Rulemakings and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and it is requested that copies be transmitted either by means of facsimile transmission to (301) 415–3725 or by e-mail to *OGCMailCenter@nrc.gov*. A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the ADAMS Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov*.

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

Date of amendment request: June 14, 2006.

Description of amendment request: The proposed amendment will permit Millstone Power Station, Unit 3 a one-time, 5-year extension, to Type A testing, of a surveillance requirement referenced in Technical Specification (TS) 4.6.1, relevant to the containment structure. TS 4.6.1 specifies performance of an integrated leak rate test at a frequency of up to 10 years with allowance for a 15-month extension.

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

Criterion 1:

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

Response: No.

The proposed extension to Type A testing cannot increase the probability of an accident previously evaluated since extension of the containment Type A testing is not a physical plant modification that could alter the probability of accident occurrence nor, is it an activity or modification that by itself could lead to equipment failure or accident initiation.

The proposed one-time, five-year extension to Type A testing does not result in a significant increase in the consequences of an accident as documented in NUREG–1493. The NUREG notes that very few potential containment leakage paths are not identified by Type B and C tests. It concludes that even reducing the Type A (ILRT) testing frequency to once per twenty years leads to an imperceptible increase in risk.

DNC provides a high degree of assurance through indirect testing and inspection that the containment will not degrade in a manner detectable only by Type A testing. The last two Type A tests identified containment leakage within acceptance criteria, indicating a very leak-tight containment. Inspections required by the ASME Code [American Society of Mechanical Engineers Boiler and Pressure Vessel Code] are also performed in order to identify indications of containment degradation that could affect leak-tightness. Separately, Type B and C testing required by Technical Specifications, identifies any containment opening from design penetrations, such as valves, that would otherwise be detected by a Type A test. These factors establish that a one-time, five-year extension to the Millstone Power Station Unit 3 Type A test interval will not represent a significant increase in the consequences of an accident.

Criterion 2:

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

Response: No.

The proposed revision to Technical Specifications adds a one-time extension to the current interval for Type A testing for Millstone Power Station Unit 3. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test. The proposed extension to Type A testing does not create the possibility of a new or different type of accident since there are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure.

Criterion 3:

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

Response: No.

The proposed revision to Millstone Power Station Unit 3 Technical Specifications adds a one-time extension to the current interval for Type A testing. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test for Millstone Power Station Unit 3. RG [Regulatory Guide] 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of CDF [core damage frequency] below 10^{-6}/yr and increases in LERF [large early release frequency] below 10^{-7}/yr. Since the ILRT [integrated leak rate testing] does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from a once-per-ten-years to a once-per-fifteen-years is 3.1×10^{-7}/yr, based on internal events. RG 1.174 states that when the calculated increase in LERF is in the range of 10^{-7}/yr to 10^{-6}/yr, applications will be considered if it can be shown that the total [LERF] is less than 10^{-5}/yr. Since the total LERF for the 15-year metric is 6.3×10^{-7}/yr, then the change is considered acceptable. Increasing the ILRT interval from ten to fifteen years is, therefore, considered non-risk significant and will not significantly reduce the margin of safety. The NUREG–1493 generic study of the effects of extending containment leakage testing found that a 20-year interval in Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG–1493 generically concludes that the design containment leakage rate contributes about 0.1 percent of the overall risk. Decreasing the Type A testing frequency would have a minimal [e]ffect on this risk since 95% of the Type A detectable leakage paths would already be detected by Type B and C testing.

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

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

NRC Acting Branch Chief: Brooke D. Poole.

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: June 21, 2006.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3.7.3, Action a, to extend the allowed outage time (AOT) for one inoperable intake cooling water (ICW) pump from 7 days to 14 days. The proposed amendments were prepared in accordance with the guidance provided by the NRC in Regulatory Guide 1.174, ''An Approach for Using Probabilistic Risk Assessment in Risk Informed Decisions on Plant-Specific Changes to the Licensing Basis'' and Regulatory Guide 1.177, ''An Approach for Plant-Specific, Risk-Informed Decisionmaking; Technical Specifications.''

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 affects the AOT for TS 3.7.3, Action a. The proposed change allows an extension of the current AOT for an inoperable ICW pump from 7 days to 14 days. The proposed change does not affect the design of the ICW System, the operational characteristics or function of the ICW System, the interfaces between the ICW System and other plant systems, or significantly affect the reliability of the ICW System. Limiting conditions for operation and their associated allowed outage times are not considered initiating conditions for any accident previously evaluated, nor is the ICW System considered an initiator for any accident previously evaluated. The ICW System provides the cooling water to the safety related CCW [component cooling water] heat exchangers. The ICW System also provides cooling water to the TPCW [turbine plant cooling water] heat exchangers and supplies water to the Lube Water System. During accident conditions, the ICW System performs the accident mitigation function of removing the heat load from the CCW System to support both reactor heat removal and containment heat removal requirements. The consequences of accidents previously evaluated are not affected by the proposed change in AOT. To fully evaluate the effect of the proposed ICW AOT extension, PRA [probabilistic risk assessment] methods and a deterministic analysis were utilized. The results of the analysis show no significant increase in Core Damage Frequency or Large Early Release Frequency based upon the guidance provided in Regulatory Guides 1.174 and 1.177.

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 probability of a new or different accident from any accident previously evaluated?

Response: No. The proposed change does not involve a change in the design, configuration, or method of operation of the plant. The proposed change will not alter the manner in which equipment operation is initiated, nor will the functional demands on credited equipment be changed. The proposed change allows operation of a Turkey Point unit to continue while an ICW pump is repaired and tested. The proposed extension does not affect the interaction of an ICW pump with any system whose failure or malfunction can initiate an accident. As such, no new failure modes are being introduced.

Therefore, the proposed action 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 alter the plant design, nor does it affect the assumptions contained in the safety analyses. Specifically, there are no changes being made to the ICW design, including instrument setpoints. The proposed change has been evaluated both deterministically, and using risk-informed methods. Based upon these evaluations, margins of safety ascribed to ICW availability and to plant risk have been determined to not be significantly reduced. The evaluation has concluded the following with respect to the proposed change:

Applicable regulatory requirements will continue to be met, adequate defense-in-depth will be maintained, sufficient safety margins will be maintained, and any increases in CDF [core damage frequency] and LERF [large early release frequency] are small and consistent with the NRC Safety Goal Policy Statement (**Federal Register**, Vol. 5.1, P. 30028 (51 FR 30028), August 4, 1986) as interpreted by NRC Regulatory Guides 1.174 and 1.177. Furthermore, increases in risk posed by potential combinations of equipment out of service during the proposed extended ICW pump AOT will be managed under a configuration risk management program consistent with 10 CFR 50.65, ''Requirements for monitoring the effectiveness of maintenance at nuclear power plants,'' paragraph (a)(4).

The availability of the other ICW pumps and the use of on-line risk assessment tools provide adequate compensation for the potential small incremental increase in plant risk associated with the extended ICW pump AOT.

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

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

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

NRC Branch Chief: L. Raghavan.

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

Date of amendment request: July 6, 2006.

Description of amendment request: The proposed amendments would incorporate new large-break loss-of-coolant accident (LBLOCA) analyses using the realistic LBLOCA methodology in the NRC-approved WCAP–16009–P–A, ''Realistic Large Break LOCA [loss-of-coolant-accident] Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM),'' and would revise Technical Specification (TS) 5.6.5.b to include reference to WCAP–16009–P–A.

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

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

Response: No.

This license amendment request proposes to incorporate large break loss of coolant accident analyses using the ASTRUM methodology, documented in WCAP–16009–P–A, ''Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)'', in the Prairie Island Nuclear Generating Plant licensing basis and add reference to WCAP–16009–P–A in the Technical Specification's list of approved methodologies for establishing core operating limits.

Accident analyses are not accident initiators, therefore, this proposed licensing basis change does not involve a significant increase in the probability of an accident. The analyses using ASTRUM demonstrated that the acceptance criteria in 10 CFR 50.46, ''Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors'', were met. The NRC has approved WCAP–16009–P–A for application to two-loop Westinghouse plants with upper plenum injection. Since the Prairie Island Nuclear Generating Plant is a two-loop Westinghouse plants with upper head injection and the analysis results meet the 10 CFR 50.46 acceptance criteria, this change does not involve a significant increase in the consequences of an accident.

Addition of the reference to WCAP–16009–P–A in the Technical Specifications is an administrative change that does not affect the probability or consequences of an accident previously evaluated.

The changes proposed in this license amendment do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

This license amendment request proposes to incorporate large break loss of coolant accident analyses using the ASTRUM methodology, documented in WCAP–16009–P–A, ''Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)'', in the Prairie Island Nuclear Generating Plant licensing basis and add reference to WCAP–16009–P–A in the Technical Specification's list of approved methodologies for establishing core operating limits.

There are no physical changes being made to the plant as a result of using the Westinghouse ASTRUM analysis methodology in WCAP–16009–P–A for performance of the large break loss of coolant accident analyses. No new modes of plant operation are being introduced. The configuration, operation and accident response of the structures or components are unchanged by utilization of the new analysis methodology. Analyses of transient events have confirmed that no transient event results in a new sequence of events that could lead to a new accident scenario. The parameters assumed in the analysis are within the design limits of existing plant equipment.

In addition, employing the Westinghouse ASTRUM large break loss of coolant accident analysis methodology does not create any new failure modes that could lead to a different kind of accident. The design of all systems remains unchanged and no new equipment or systems have been installed which could potentially introduce new failure modes or accident sequences. No changes have been made to any reactor protection system or emergency safeguards features instrumentation actuation setpoints.

Based on this review, it is concluded that no new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed methodology changes.

Addition of the reference to WCAP–16009–P–A in the Technical Specifications is an administrative change that does not create the possibility of a new or different kind of accident.

The licensing basis and Technical Specification changes proposed in this license amendment do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

This license amendment request proposes to incorporate large break loss of coolant accident analyses using the ASTRUM methodology, documented in WCAP–16009–P–A, ''Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)'', in the Prairie Island Nuclear Generating Plant licensing basis and add reference to WCAP–16009–P–A in the Technical Specification's list of approved methodologies for establishing core operating limits.

The analyses using ASTRUM demonstrated that the applicable acceptance criteria in 10 CFR 50.46, ''Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors'' are met. Margins of safety for large break loss of coolant accidents include quantitative limits for fuel performance established in 10 CFR 50.46. These acceptance criteria and the associated margins of safety are not being changed by this proposed new methodology. The NRC has approved WCAP–16009–P–A for application to two-loop Westinghouse plants with upper head injection. Since the Prairie Island Nuclear Generating Plant is a two-loop Westinghouse plants with upper plenum injection and the analysis results meet the 10 CFR 50.46 acceptance criteria, this change does not involve a significant reduction in a margin of safety.

Addition of the reference to WCAP–16009–P–A in the Technical Specifications is an administrative change that does not involve a significant reduction in a margin of safety.

Addition of the reference to WCAP–16009–P–A in the Technical Specifications is an administrative change that does not involve a significant reduction in a margin of safety.

The licensing basis and Technical Specification changes proposed in this license amendment 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 requests involve 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 Acting Branch Chief: Martin Murphy.

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

Date of amendment request: August 11, 2006.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to relocate response time limit tables for the reactor trip system and engineered safety features actuation system to the Updated Final Safety Analysis Report.

The August 11, 2006, application supersedes the previous application related to relocation of response time limits, dated August 19, 2005, which was noticed in the **Federal Register** on December 20, 2005 (70 FR 75496). Instead of changing the response time definitions in TSs 1.12 and 1.26, as proposed in the August 19, 2005, application, the licensee proposes to revise certain TS Bases to clarify that Nuclear Regulatory Commission approval would be required to use a means other than testing to verify that response times are within limits.

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

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

Response: No.

The proposed amendment relocates the instrument response time limits for the reactor trip system (RTS) and engineered safety feature actuation system (ESFAS) from the technical specifications to the Updated Final Safety Analysis Report (UFSAR). The proposed amendment conforms to the guidance given in Enclosures 1 and 2 of Generic Letter 93–08. Neither the response time limits nor the surveillance requirements for performing response time testing will be altered by this submittal. The overall RTS and ESFAS functional capabilities will not be changed and assurance that action requirements of the reactor trip and engineered safety features systems are completed within the time limits assumed in the accident analyses is unaffected by the proposed amendment.

Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed amendment will not change the physical plant or the modes of plant operation defined in the operating license. The change does not involve the addition or modification of equipment nor does it alter the design or operation of plant systems.

Therefore, operation of the facility in accordance with the proposed amendment will 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 measurement of instrumentation response times at the frequencies specified in the technical specification provides assurance that actions associated with the reactor trip and engineered safety features are accomplished within the time limits assumed in the accident analyses. The response time limits and the measurement frequencies remain unchanged by the proposed amendment.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions.

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 Acting Branch Chief: Brooke D. Poole.

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: July 14, 2006.

Description of amendment requests: This amendment application proposes to delete duplicative notifications, reporting, and restart requirements if a safety limit is violated; replace plant-specific position titles with generic position titles; and make several additional administrative changes.

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

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

Response: No.

The proposed change to remove the duplicative safety limit reporting requirements from the TSs [Technical Specifications] does not affect the plant or operation of the plant. The change simply removes duplicative information from the TSs that is covered in the NRC regulations. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to make plant-specific position/organizational titles more generic do not affect any plant structures, systems, and components, and have no effect on plant operations. The proposed changes are administrative and do not affect any existing limits. Accident initial conditions, probability, and assumptions remain as previously analyzed. The proposed changes will have no effect on accident initiation frequency. The proposed changes do not invalidate the assumptions used in evaluating the radiological consequences of any accident. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The remaining changes are administrative and do not modify the qualifications, responsibilities, or requirements for the positions. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change to remove the duplicative safety limit reporting requirements from the TSs does not introduce any new accident scenarios, failure mechanisms, or limiting single failures. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed change has no adverse effect on any safety-related system or component and does not challenge the performance or integrity of any safety related system. This change is considered an administrative action to remove duplicative reporting requirements. Therefore, this proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes to make plant-specific position/organizational titles more generic are administrative and do not introduce any new or different accident initiators. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The remaining proposed changes are administrative and do not modify the qualifications, responsibilities, or requirements for the positions. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed changes are administrative and do not involve any reduction in a margin of safety. Removal of duplicative information, replacing plant-specific position titles with generic position titles, and the other proposed administrative changes do not affect compliance with the regulations. 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 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.

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: July 14, 2006.

Description of amendment requests: The proposed change incorporates a description of the parent tube inspection limitation adjacent to the nickel band portion of the lower sleeve joint and provides the basis for the structural and leakage integrity of the joint being ensured with the existing inspection of the parent tube adjacent to the nickel band region.

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 revises the San Onofre Units 2 and 3 Technical Specifications (TS) Section 5.5.2.11.f.1.j to provide a description of the parent tube inspection limitation adjacent to the nickel band and to provide the basis for the structural and leakage integrity. This is supported by Westinghouse Topical Report SG–SGDA–05–48–P Revision 1, ''WOG [Westinghouse Owners Group] PA–MSC–0190, Revision 1: Test Results Related to TIG [tungsten inert gas] and Alloy 800 Sleeve Installation in ¾ Inch and ⅞ Inch OD SG [steam generator] Tubing In-Service Inspection Requirements.''

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.

Steam generator tube leakage and structural integrity will be maintained during all plant conditions upon implementation of the proposed inspection scope and repair limit changes to the San Onofre Units 2 and 3 Technical Specifications. This change does not introduce any new mechanisms that might result in a different kind of accident from those previously evaluated.

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.

Structural and leakage integrity of the steam generator sleeve joint is ensured with the existing inspection of the parent tube adjacent to the nickel band region.

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.

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

Date of amendment request: February 28, 2006.

Brief description of amendments: This request proposes changes to Technical Specification (TS) 3/4.8.2.1, ''DC Sources—Operating,'' and 3/4.8.2.2, ''DC Sources—Shutdown,'' and the addition of a new TS 3/4.8.2.3, ''Battery Parameters.''

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

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

The proposed change rearranges the Technical Specifications for the direct current electrical power system, and adds new Conditions and required actions with revised completion times to allow for battery charger inoperability. Neither the direct current electrical power subsystem nor associated battery chargers are initiators of an accident sequence previously evaluated. Performance of plant operational activities in accordance with the proposed Technical Specification changes ensures that the direct current electrical power subsystem is capable of performing its function as previously described. Therefore, the mitigating functions supported by the subject subsystem will continue to provide the protection assumed by the safety analysis.

Relocation of preventive maintenance surveillances and certain operating limits and actions to a ''Battery Monitoring and Maintenance Program'' will not challenge the ability of the subject subsystem to perform its design function. Maintenance and monitoring currently required will continue to be performed. In addition, the direct current electrical power subsystem is within the scope of 10 CFR 50.65, ''Requirements for monitoring the effectiveness of maintenance at nuclear power plants,'' which will ensure continued control of maintenance activities associated with the subject subsystem.

Revision of battery performance test interval to 12 months from 18 months in 4.8.2.1.f (now 4.8.2.3.f.1) is a conservative change that is intended to ensure continued battery operability. In addition, a surveillance requirement will be added as 4.8.2.3.f.2 to require performance discharge tests at least once per 24 months for any battery reaching 85% of the service life expected for the application and capacity is equal to or greater than 100% of the manufacturer's rating. Surveillance requirement 4.8.2.3.f.2 is an additional criterion that supplements 4.8.2.3.f.1. Modified performance tests of batteries that have reached 85% of their service life are to be performed at 12-month intervals with capacity less [than] 100% of the manufacturer's rating, and at 24-month intervals if the capacity is 100% or greater. These surveillance requirements are consistent with the requirements of IEEE–450.

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

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

The proposed change does not involve any physical alteration of the units. No new equipment is introduced, and installed equipment is not operated in a new or different manner. The proposed changes do not affect setpoints for initiation of protective or mitigating actions.

Revision of battery performance test interval to 12 months from 18 months in 4.8.2.1.f (now 4.8.2.3.f.1) is a conservative change that is intended to ensure continued battery operability. In addition, a surveillance requirement will be added as 4.8.2.3.f.2 to require performance discharge tests at least once per 24 months for any battery reaching 85% of the service life expected for the application and capacity is equal to or greater than 100% of the manufacturer's rating. Surveillance requirement 4.8.2.3.f.2 is an additional criterion that supplements 4.8.2.3.f.1. Modified performance tests of batteries that have reached 85% of their service life are to be performed at 12-month intervals with capacity less [than] 100% of the manufacturer's rating, and at 24-month intervals if the capacity is 100% or greater. These surveillance requirements are consistent with the requirements of IEEE–450.

Operability of the DC [direct currrent] electrical power subsystems in accordance with the proposed technical specifications is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant.

The proposed changes will not alter the manner in which equipment operation is initiated, nor will the functional demands on credited equipment be changed. No alteration in the operating procedures is proposed, and no change is being made to procedures relied upon in response to an off-normal event. No new failure modes are being introduced, and the proposed change does not alter assumptions made in the safety analyses.

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

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

The proposed change will not adversely affect operation of plant equipment and will not result in a change to the setpoints at which protective actions are initiated. Sufficient DC capacity to support operation of mitigation equipment is ensured. The provisions of the Battery Monitoring and Maintenance Program will ensure that the station batteries are maintained in a highly reliable manner.

Revision of battery performance test interval to 12 months from 18 months in 4.8.2.1.f (now 4.8.2.3.f.1) is a conservative change that is intended to ensure continued battery operability. In addition, a surveillance requirement will be added as 4.8.2.3.f.2 to require performance discharge tests at least once per 24 months for any battery reaching 85% of the service life expected for the application and capacity is equal to or greater than 100% of the manufacturer's rating. Surveillance requirement 4.8.2.3.f.2 is an additional criterion that supplements 4.8.2.3.f.1. Modified performance tests of batteries that have reached 85% of their service life are to be performed at 12-month intervals with capacity less [than] 100% of the manufacturer's rating, and at 24-month intervals if the capacity is 100% or greater. These surveillance requirements are consistent with the requirements of IEEE–450.

The equipment fed by the DC electrical system will continue to provide adequate power to safety-related loads in accordance with analysis assumptions.

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: A. H. Gutterman, Esq., Morgan, Lewis & Bockius, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Branch Chief: David Terao.

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

Date of amendment request: June 7, 2006.

Brief description of amendments: The proposed change would revise the Spent Fuel Pool (SFP) and In-Containment Storage Area Criticality Analysis as described in Section 5.6 of the Technical Specifications (TSs).

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.

There is no increase in the probability of an accident. The proposed change does allow a greater number of fuel storage configurations in SFP. While this could increase the probability of a fuel misloading, the presence of sufficient soluble boron in the SFP precludes criticality as a result of the misloading. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures and will be in accordance with the TS and the spent fuel rack storage configuration limitations of UFSAR [updated final safety analysis report] Chapter 9.1.2.

Reactivity changes due to SFP temperature changes have been evaluated. The base case criticality analysis covers a ''normal'' SFP temperature range of 50 °F to 160 °F. Spent fuel pool temperature accidents are considered outside the normal temperature range extending from 50 °F to 240 °F. In all SFP temperature accident cases, sufficient reactivity margin is available to the 0.95 k_{eff} limit without requiring additional soluble boron above the base case level. Because adequate soluble boron will be maintained in the SFP water to maintain $k_{eff} < 0.95$, the consequences of a loss of normal cooling to the SFP will not be increased.

There is no increase in the consequences of the accidental misloading of spent fuel assemblies into the SFP racks. The criticality analysis demonstrates that the pool k_{eff} will remain ≤ 0.95 following an accidental misloading due to the boron concentration of the pool. The current TS limitation will ensure that an adequate SFP boron concentration is maintained.

The criticality analysis shows the consequences of a fuel assembly drop accident in the SFP are not affected when considering the presence of soluble boron. The rack k_{eff} remains ≤ 0.95.

The editorial changes proposed in this license amendment request do not impact the probability or consequences of an accident.

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

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

Response: No.

Spent fuel handling accidents are not new or different types of accidents; they have been analyzed in Section 15.7.4 of the UFSAR.

Criticality accidents in the SFP are not new or different types of accidents. They have been analyzed in the UFSAR and in Criticality Analysis Reports associated with specific licensing amendments for fuel enrichments that are assumed for the proposed change. Because the proposed SFP storage configuration limitations will be similar to the current ones, the new limitations will not have any significant effect on normal SFP operations and maintenance, and will not create any possibility of a new or different kind of accident. Verifications will continue to be performed to ensure that the SFP loading configuration meets specified requirements.

The misloading of a fuel assembly in the required storage configuration has been evaluated. In all cases, the rack k_{eff} remains ≤ 0.95. Removal of an RCCA [rod cluster control assembly] from a checkerboard storage configuration has been analyzed and found to be bounded by the misloading of a fuel assembly.

As discussed above, the proposed changes will not create the possibility of a new or different kind of accident. There is no significant change in plant configuration, equipment design, or equipment.

The editorial changes proposed in this license amendment request do not impact the design basis accidents of STP [South Texas Project].

Under the proposed amendment, no changes are being made to the racks themselves, to any other systems, or to the physical structures of the Fuel Handling Building.

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 TS changes and the resulting spent fuel storage operation limits will provide [an] adequate safety margin to ensure that the stored fuel assembly array always remains subcritical. Those limits are based on a plant-specific criticality analysis performed in accordance with Westinghouse spent fuel rack criticality analysis methodology.

While the criticality analysis utilized credit for soluble boron, storage configurations have been defined using 95/95 k_{eff} calculations to ensure that the spent fuel rack k_{eff} is < 1.0 with no soluble boron. Soluble boron credit is used to offset uncertainties, tolerances, and off-normal conditions, and to provide subcritical margin such that the SFP k_{eff} is maintained ≤ 0.95.

The loss of substantial amounts of soluble boron from the SFP that could lead to k_{eff} exceeding 0.95 has been previously evaluated and approved (Ref. 4 and 5) and shown to be not credible. A safety evaluation has been performed which shows that dilution of the SFP boron concentration from 2500 ppm [part per million] to 700 ppm is not credible. Also, the spent fuel rack k_{eff} will remain < 1.0 (with a 95/95 confidence level) with the SFP flooded with unborated water. These safety analyses demonstrate a level of safety comparable to the conservative criticality analysis methodology required by Westinghouse WCAP–14416–P–A.

The editorial changes proposed in this license amendment request do not affect the margin of safety.

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

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

Attorney for licensee: A. H. Gutterman, Esq., Morgan, Lewis & Bockius, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Branch Chief: David Terao.

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

Date of amendment request: March 31, 2006.

Brief description of amendments: The amendments requested would revise Technical Specifications (TS) requirement 5.0, ''ADMINISTRATIVE CONTROLS.''

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

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

Response: No.

The proposed change involves organizational changes at the executive level and does not impact nor effect accident analysis assumptions. The method and tools used to maintain, and produce proposed changes to, the Technical Specifications has no bearing on any accident analysis assumptions. Therefore, these assumptions are preserved and there is no change in the probability or consequences of any previously evaluated accident.

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

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

Response: No.

The proposed change involves an organizational change due to a change in title. There are no changes in existing reporting relationships or assigned responsibilities for safe operation of CPSES. The proposed re-issuance of the entire Technical Specifications stems from a change in the software utilized by TXU Power to produce and maintain the Technical Specifications. This software is not used to operate the plant nor is it used to establish any operational limits.

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The proposed change will not effect the normal method of plant operation. No performance requirements will be affected or eliminated. The proposed change will not result in physical alteration to any plant system nor will there be any change in the method by which any safety-related plant system performs its safety function.

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

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

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

Response: No.

The proposed changes do 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.

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

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

Attorney for licensee: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Branch Chief: David Terao.

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

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

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

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

Date of amendment request: July 20, 2006.

Brief description of amendment request: The proposed amendment would revise the Vogtle Electric Generating Plant (VEGP), Units 1 and 2, Technical Specifications (TS) 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' to incorporate changes in the SG inspection scope for VEGP, Unit 1 during Refueling Outage 13 and the subsequent operating cycle, and VEGP Unit 2 during Refueling Outage 12 and the subsequent operating cycle. The proposed changes modify the inspection requirements for portions of SG tubes within the tubesheet region of the SGs.

Date of publication of individual notice in ***Federal Register:*** July 31, 2006 (71 FR 43225).

Expiration date of individual notice: 30-day, August 30, 2006; 60-day, September 29, 2006.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

AmerGen Energy Company, LLC, Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of application for amendment: February 2, 2005, as supplemented by letters dated April 19, April 21, and June 13, 2006.

Brief description of amendment: The amendment revised the Oyster Creek Nuclear Generating Station Technical Specifications (TSs) to incorporate the isolation trip setting and the instrumentation surveillance requirements of the reactor water clean-up system high energy line break detection and isolation equipment.

Date of Issuance: August 25, 2006.

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

Amendment No.: 259.

Facility Operating License No. DPR–16: The amendment revised the TSs.

Date of initial notice in ***Federal Register:*** March 15, 2005 (70 FR 12744). The April 19, April 21, and June 13, 2006, letters provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated August 25, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 14, 2004.

Brief description of amendment: The amendment deleted redundant administrative responsibilities, changed certain administrative titles and included editorial corrections and clarifications.

Date of issuance: August 9, 2006.

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

Amendment No.: 223.

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

Operating License and Technical Specifications.

Date of initial notice in ***Federal Register:*** March 1, 2005 (70 FR 9990)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 9, 2006.

No significant hazards consideration comments received: No

Entergy Operations, Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: March 15, 2005, as supplemented by letters dated March 22, June 2, and July 12, 2006.

Brief description of amendment: The amendment modified Technical Specification (TS) 6.5.9, ''Steam Generator (SG) Program,'' and TS 6.9.1.5, ''Steam Generator Tube Inspection Report,'' to eliminate the need to inspect a portion of the tube within the SG tubesheet region, thereby potentially allowing flaws to remain in the uninspected region.

Date of issuance: August 29, 2006.

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

Amendment No.: 207.

Facility Operating License No. NPF–38: The amendment revised the Technical Specifications and the Facility Operating License.

Date of initial notice in ***Federal Register:*** June 21, 2005 (70 FR 35737). The March 22, June 2, and July 12, 2006, supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: October 25, 2005.

Brief description of amendment: The amendment modified the Surveillance Requirements related to Waterford 3 Technical Specification 3.1.1.3, ''Moderator Temperature Coefficient,'' to permit use of the Startup Test Activity Reduction Program (WCAP–16011–P–A).

Date of issuance: August 29, 2006.

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

Amendment No.: 206.

Facility Operating License No. NPF–38: The amendment revised the Technical Specifications and the Facility Operating License.

Date of initial notice in ***Federal Register:*** December 6, 2005 (70 FR 72673). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 29, 2006.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of application for amendments: February 27, 2004, as supplemented by letters dated October 25, 2004, October 10, 2005, April 27, May 30, June 16, and August 4, 2006.

Brief description of amendments: This amendment incorporated a revision to the Technical Specifications (TSs) and licensing and design bases that supports a full-scope application of an Alternative Source Term methodology.

Date of issuance: August 23, 2006.

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

Amendment Nos. 185, 146.

Facility Operating License Nos. NPF–39 and NPF–85. This amendment revised the TSs.

Date of initial notice in ***Federal Register:*** June 22, 2004 (69 FR 34700). The supplements provided clarifying information that did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as originally published in the **Federal Register**. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 23, 2006.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit No. 2, Beaver County, Pennsylvania

Date of application for amendment: October 14, 2005, as supplemented March 31, 2006.

Brief description of amendment: The amendment revised Technical Specifications 3/4 8.2.3 and 3/4 8.2.4 to permit implementation of design changes associated with a battery charger upgrade during the fall 2006 refueling outage.

Date of issuance: August 28, 2006.

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

Amendment No: 157.

Facility Operating License No. NPF–73. Amendment revised the License and the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 22, 2005 (70 FR 70642). The supplement dated March 31, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

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

No significant hazards consideration comments received: No.

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 application for amendments: February 16, 2006, as supplemented by letters dated May 11 and July 13, 2006.

Brief description of amendments: The amendments revised the Technical Specification (TS) requirements related to steam generator tube integrity consistent with NRC-approved Revision 4 to TS Task Force (TSTF) Standard Technical Specification Change Traveler TSTF–449, ''Steam Generator Tube Integrity.''

Date of issuance: August 22, 2006.

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

Amendment Nos.: 223 and 229.

Renewed Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 11, 2006 (71 FR 18374). The supplements dated May 11 and July 13, 2006, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 22, 2006.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 11, 2005, as revised by letter dated November 8, 2005, as supplemented by letter dated April 12, 2006.

Brief description of amendment: The amendment revised TS 4.2.1, ''Fuel Assemblies,'' to permit the use of AREVA (Framatome ANP) M5 advanced alloy for fuel rod cladding and structural components such as guide tubes, intermediate spacer grids, end plugs, and guide thimble tubes at the Fort Calhoun Station, Unit 1 (FCS). M5 will be used beginning with Refueling Cycle 24. The M5 cladding is a proprietary zirconium-based alloy that is chemically different from that of zircaloy and ZIRLO, the fuel cladding materials currently approved for use in the FCS TS. In addition, TS 5.9, ''Reporting Requirements,'' was revised to include the Framatome ANP topical report evaluating the impact of M5 material properties on NRC-approved methodologies used at the FCS.

Date of issuance: August 30, 2006.

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

Amendment No.: 241.

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

Date of initial notice in ***Federal Register:*** December 6, 2005 (70 FR 72675). The April 12, 2006, supplemental letter provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a safety evaluation dated August 30, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 23, 2006.

Brief description of amendment: The amendment revised Paragraph 2.C.(6) of the facility operating license to clarify that the license condition that limits the number of fuel assemblies that can be outside of approved shipping containers, fuel storage racks, or the reactor does not apply to fuel assemblies stored in approved dry spent fuel storage systems.

Date of issuance: August 28, 2006.

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

Amendment No.: 169.

Facility Operating License No. NPF–57: This amendment revised Paragraph 2.C.(6).

Date of initial notice in ***Federal Register:*** May 9, 2006 (71 FR 27003).

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

No significant hazards consideration comments received: No.

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

Dates of application for amendments: March 29, 2006, as supplemented on June 5, 2006.

Brief description of amendments: The amendments revised the Technical Specification (TS) requirements related to steam generator tube integrity. The changes are consistent with Nuclear Regulatory Commission (NRC)-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.'' The 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.

Date of issuance: August 28, 2006.

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

Amendment Nos.: 144 and 124.

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 23961). The supplement dated June 5, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the **Federal Register** on April 25, 2006 (71 FR 23961).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 30, 2005 (TS–05–02).

Brief description of amendments: The amendment revises Technical Specification (TS) Section 5.0 ''Design Features,'' to conform with NUREG–1431, Revision 3, ''Standard Technical Specifications for Westinghouse Plants.'' The changes include elimination of the exclusion area, low population zone, and effluent subsections and associated figures referred to in Section 5.1 ''Site;'' elimination of Section 5.2 ''Containment;'' elimination of Section 5.4 ''Reactor Coolant System,'' as well as Section 5.5 ''Meteorological Tower Location,'' and its figure. Lastly, a change has been made to TS Section 6.0, Administrative Control,'' to acquire the component cyclic or transient limits currently located in the ''Design Features'' section.

Date of issuance: August 2, 2006.

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

Amendment Nos.: 309 and 298.

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

Date of initial notice in ***Federal Register:*** November 8, 2005 (70 FR 67752).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 14, 2005 (TS–05–07), as supplemented by letter dated March 31, 2006.

Brief description of amendment: The amendment revises Technical Specification (TS) Section 5.7.2.19, ''Containment Leakage Rate Testing Program,'' to allow a one time, 5-year extension to the current 10 year test interval for the performance-based leakage rate test program for 10 CFR 50, Appendix J, Type A tests.

Date of issuance: August 22, 2006.

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

Amendment No.: 63.

Facility Operating License No. NPF–90: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 28, 2006 (71 FR 10078). The supplemental letter provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a

Safety Evaluation dated August 22, 2006.

No significant hazards consideration comments received: No.

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

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

Brief description of amendment: The amendment added a new Technical Specification (TS) 3.1.9, ''RCS [Reactor Coolant System] Boron Limitations < 500 °F,'' and revised TS 3.3.1, ''Reactor Trip System (RTS) Instrumentation,'' for the power range neutron flux—low reactor trip function.

Date of issuance: August 21, 2006.

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

Amendment No.: 174.

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

*Date of initial notice in **Federal Register:*** May 23, 2006 (71 FR 29682). The supplemental letter dated December 21, 2005, provided clarifying information that did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination published in the **Federal Register**.

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

No significant hazards consideration comments received: No.

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

Date of amendment request: February 7, 2006, as supplemented by letter dated July 25, 2006.

Brief description of amendment: The amendment revised Technical Specification Table 3.3.1–1, ''Reactor Trip System Instrumentation,'' by adding the existing Surveillance Requirement 3.3.1.16 to Function 3.a of the table.

Date of issuance: August 29, 2006.

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

Amendment No.: 165.

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

*Date of initial notice in **Federal Register:*** February 28, 2006 (71 FR 10080). The supplemental letter dated July 25, 2006, provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination published in the **Federal Register**.

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

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 1st day of September 2006.

For the Nuclear Regulatory Commission.

Timothy McGinty,

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

[FR Doc. E6–14938 Filed 9–11–06; 8:45 am]

BILLING CODE 7590–01–P

OVERSEAS PRIVATE INVESTMENT CORPORATION

September 21, 2006 Board of Directors Meeting

TIME AND DATE: Thursday, September 21, 2006, 10 a.m. (Open Portion), 10:15 a.m. (Closed Portion).

PLACE: Offices of the Corporation, Twelfth Floor Board Room, 1100 New York Avenue, NW., Washington, DC.

STATUS: Meeting Open to the Public from 10 a.m. to 10:15 a.m. Closed portion will commence at 10:15 a.m. (approx.).

MATTERS TO BE CONSIDERED:

1. President's Report.
2. Approval of July 13, 2006 Minutes (Open Portion).

FURTHER MATTERS TO BE CONSIDERED: (Closed to the Public 10:15 a.m.)

1. Report from Audit Committee.
2. Proposed FY2008 Budget.
3. Finance Project—Latin and Central America.
4. Finance Project—Global.
5. Approval of July 13, 2006 Minutes (Closed Portion).
6. Pending Major Projects.
7. Reports.

FOR FURTHER INFORMATION CONTACT: Information on the meeting may be obtained from Connie M. Downs at (202) 336–8438.

Dated: September 8, 2006.

Connie M. Downs,

Corporate Secretary, Overseas Private Investment Corporation.

[FR Doc. 06–7611 Filed 9–8–06; 11:53 am]

BILLING CODE 3210–01–M

SECURITIES AND EXCHANGE COMMISSION

Sunshine Act Meeting

Notice is hereby given, pursuant to the provisions of the Government in the Sunshine Act, Pub. L. 94–409, that the Securities and Exchange Commission will hold the following meeting during the week of September 11, 2006:

A Closed Meeting will be held on Tuesday, September 12, 2006 at 10 a.m.

Commissioners, Counsels to the Commissioners, the Secretary to the Commission, and recording secretaries will attend the Closed Meeting. Certain staff members who have an interest in the matters may also be present.

The General Counsel of the Commission, or his designee, has certified that, in his opinion, one or more of the exemptions set forth in 5 U.S.C. 552b(c)(3), (5), (7), (9)(B) and (10) and 17 CFR 200.402(a) (3), (5), (7), (9)(ii), and (10) permit consideration of the scheduled matters at the Closed Meeting.

Commissioner Atkins, as duty officer, voted to consider the items listed for the closed meeting in closed session and determined that no earlier notice thereof was possible.

The subject matters of the Closed Meeting scheduled for Tuesday, September 12, 2006 will be: Formal orders of investigation; institution and settlement of injunctive actions; institution and settlement of administrative proceedings of an enforcement nature; adjudicatory matters; and other matters related to enforcement proceedings.

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: September 8, 2006.

Jill M. Peterson,

Assistant Secretary.

[FR Doc. 06–7607 Filed 9–8–06; 11:09 am]

BILLING CODE 8010–01–P