Contact person for more information: Michelle Schroll, (301) 415–1662.

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

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This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, of 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: June 1, 2006.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 06–5163 Filed 6–2–06; 10:21 am]

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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 May 12, 2006 to May 24, 2006. The last biweekly notice was published on May 23, 2006 (71 FR 29671).

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.*

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

Date of amendments request: April 26, 2006.

Description of amendment request: The proposed amendment would modify technical specification (TS) requirements for inoperable snubbers by adding Limiting Condition for Operation 3.0.8. The changes are consistent with Nuclear Regulatory Commission approved Industry/Technical Specification Task Force (TSTF) standard TS change TSTF–372, Revision 4.

The NRC staff issued a notice of availability of a model safety evaluation and model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on May 4, 2005 (70 FR 23252). The licensee affirmed the applicability of the model NSHC determination in its application dated April 26, 2006.

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

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

Response: No.

The proposed change allows a delay time before declaring supported TS systems inoperable when the associated snubber(s) cannot perform its required safety function. Entrance into Actions or delaying entrance into Actions is not an initiator of any accident previously evaluated.

Consequently, the probability of an accident previously evaluated is not significantly increased. The consequences of an accident while relying on the delay time allowed before declaring a TS supported system inoperable and taking its Conditions and Required Actions are no different than the consequences of an accident under the same plant conditions while relying on the existing TS supported system Conditions and Required Actions.

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

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

Response: No.

The proposed change allows a delay time before declaring supported TS systems inoperable when the associated snubber(s) cannot perform its required safety function. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operations. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change allows a delay time before declaring supported TS systems inoperable when the associated snubber(s) cannot perform its required safety function. The proposed change restores an allowance in the pre-ISTS conversion TS that was unintentionally eliminated by the conversion. The pre-ISTS TS were considered to provide an adequate margin of safety for plant operation, as does the post-ISTS conversion TS. 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: David T. Conley, Associate General Counsel II—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Branch Chief: Michael L. Marshall, Jr.

Entergy Nuclear Operations, Inc., Docket No. 50–271, Vermont Yankee Nuclear Power Station (VYNPS), Vernon, Vermont

Date of amendment request: April 22, 2006.

Description of amendment request: The proposed amendment would relocate the Technical Specification (TS) requirements for shock suppressors (snubbers) to the Technical Requirements Manual (TRM) and add a new Limiting Condition for Operation (LCO) 3.0.8.

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 relocate TS 3/4.6.1 to the TRM is administrative in nature and does not involve the modification of any plant equipment or affect basic plant operation. Snubber operability and surveillance requirements will be contained in the TRM to ensure design assumptions for accident mitigation are maintained.

The proposed change to add LCO 3.0.8 allows a delay time before declaring supported TS systems inoperable when the associated snubber(s) cannot perform the required safety function. Entrance into actions or delaying entrance into actions is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The station design and safety analysis assumptions included provisions for redundancy to provide for periods when redundant systems are out-of-service per the TS. The proposed snubber LCO ensures that out-of-service time is minimized and risk is managed per 10 CFR 50.65(a)(4).

Therefore, the consequences of an accident previously evaluated are not significantly increased by this change.

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 to relocate TS 3/4.6.1 to the TRM is administrative and does not involve any physical alteration of plant equipment. The proposed change does not change the method by which any safety-related system performs its function. As such, no new or different types of equipment will be installed, and the basic operation of installed equipment is unchanged. The methods governing plant operation and testing remain consistent with current safety analysis assumptions.

[* * *]

The proposed change to add LCO 3.0.8 allows a delay time before declaring supported TS systems inoperable when the associated snubber(s) cannot perform the required safety function. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation.

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

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

Response: No.

The proposed change to relocate TS 3/4.6.1 to the TRM is administrative in nature, does not negate any existing requirement, and does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there are no changes being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. Margins of safety are unaffected by requirements that are retained, but relocated from the TS to the TRM.

[* * *]

The proposed change to add LCO 3.0.8 to TS allows a delay time before declaring supported TS systems inoperable when the associated snubber(s) cannot perform the required safety function. The proposed change retains an allowance in the current VYNPS TS while upgrading it to be more conservative for snubbers supporting multiple trains or sub-systems of an associated system. The updated TS will continue to provide an adequate margin of safety for plant operation upon incorporation of LCO 3.0.8. The station design and safety analysis assumptions provide margin in the form of redundancy to account for periods of time when system capability is reduced. This proposed change does not reduce that margin.

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

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

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

Branch Chief: Richard Laufer.

Exelon Generation Company, LLC (EGC), Docket No. 50–374, LaSalle County Station, Unit 2, LaSalle County, Illinois

Date of amendment request: April 21, 2006.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 5.5.13, ''Primary Containment Leakage Rate Testing Program,'' to reflect a one-time extension of the LaSalle County Station (LSCS), Unit 2 primary containment Type A integrated leak rate test (ILRT) date from the current requirement of no later than December 7, 2008, to prior to startup following the twelfth LSCS, Unit 2 refueling outage (L2R12).

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 changes will revise LSCS, Unit 2, TS 5.5.13, ''Primary Containment Leakage Rate Testing Program,'' to reflect a one-time extension of the primary containment Type A Integrated Leak Rate Test (ILRT) date to ''prior to startup following L2R12.'' The current Type A ILRT interval of 15 years, based on past performance, would be extended on a one-time basis by approximately 2% of the current interval.

The function of the primary containment is to isolate and contain fission products released from the reactor Primary Coolant System (PCS) following a design basis Loss of Coolant Accident (LOCA) and to confine the postulated release of radioactive material to within limits. The test interval associated with Type A ILRTs is not a precursor of any accident previously evaluated. Type A ILRTs provide assurance that the LSCS Unit 2 primary containment will not exceed allowable leakage rate values specified in the TS and will continue to perform their design function following an accident. The risk assessment of the proposed changes has concluded that there is an insignificant increase in total population dose rate and an insignificant increase in the conditional containment failure probability.

Therefore, the proposed changes do 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 changes for a one-time extension of the Type A ILRT for LSCS Unit 2 will not affect the control parameters governing unit operation or the response of plant equipment to transient and accident conditions. The proposed changes do not introduce any new equipment, modes of system operation or failure mechanisms.

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

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

Response: No.

LSCS Unit 2 is a General Electric BWR/5 plant with a Mark II primary containment. The Mark II primary containment consists of two compartments, the drywell and the suppression chamber. The drywell has the shape of a truncated cone, and is located above the cylindrically shaped suppression chamber. The primary containment is penetrated by access, piping and electrical penetrations.

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

EGC has conducted a risk assessment to determine the impact of a change to the LSCS Unit 2 Type A ILRT schedule from a baseline ILRT frequency of three times in ten years to once in 16.25 years (i.e., 15 years plus 15 months) for the risk measures of Large Early Release Frequency (i.e., LERF), Total Population Dose, and Conditional Containment Failure Probability (i.e., CCFP). This assessment indicated that the proposed LSCS ILRT interval extension has a minimal impact on public risk.

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 requested amendments involve no significant hazards consideration.

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

NRC Branch Chief: Daniel S. Collins

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

Date of amendment request: May 1, 2006.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 1.1, ''Definitions,'' TS 3.4.13, ''RCS [reactor coolant system] Operational Leakage,'' TS 5.5.8, ''Steam Generator Program,'' and add new specifications (TS 3.4.17) for ''Steam Generator (SG) Tube Integrity'' and (TS 5.6.7) for ''Steam Generator Tube Inspection Report.'' The proposed changes are necessary in order to implement the guidance for the industry initiative on Nuclear Energy Institute (NEI) 97–06, ''Steam Generator Program Guidelines.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on March 2, 2005 (70 FR 10298), on possible amendments adopting Technical Specification Task Force Change Traveller 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 1, 2006.

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

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.

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

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

The SG performance criteria proposed change to the TS identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the SG Program required by the proposed change to the TS. The program, defined by NEI 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 [kind] of accident from any accident previously evaluated.

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

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

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

NRC Branch Chief: Richard J. Laufer.

Southern 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: April 28, 2006.

Description of amendment requests: The proposed change will increase the minimum allowed boron concentration of the spent fuel pool and allow credit for soluble boron, guide tube inserts (GT-Inserts) made from borated stainless steel, and fuel storage patterns in place of Boraflex.

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.

Dropped Fuel Assembly

There is no significant increase in the probability of a fuel assembly drop accident in the spent fuel pool when assuming a complete loss of the Boraflex panels in the spent fuel pool racks and considering the presence of soluble boron in the spent fuel pool water for criticality control.

Neither the presence of soluble boron in the spent fuel pool water, nor the placement of borated stainless steel guide tube inserts (GT-Inserts) in the fuel assemblies for criticality control, will increase the probability of a fuel assembly drop accident. The handling of the fuel assemblies in the spent fuel pool has always been performed in borated water, and the quantity of Boraflex remaining in the racks or GT-Inserts placed in the fuel assemblies, has no affect on the probability of such a drop accident.

Southern California Edison (SCE) has performed a criticality analysis which shows that the consequences of a fuel assembly drop accident in the spent fuel pool are not affected when considering a complete loss of the Boraflex in the spent fuel racks and the presence of soluble boron. The rack K_{eff} remains less than or equal to 0.95.

The fuel, the fuel rack, and the fuel pool qualifications have been evaluated and determined to be unaffected by the installation of the GT-Inserts. The mechanical design configuration of the GT-Inserts is similar to the shape, size, and weight of a control element assembly (CEA) finger. Each of the GT-Inserts are approximately 0.78 inch outside diameter (OD) solid stainless steel, with a boron content of approximately 2 weight percent (w/o). A small counterbore is machined at the top for handling and a rounded bottom is machined. The OD of these GT-Inserts is less than that of a CEA finger. The material (borated stainless steel) is American Society for Testing and Materials (ASTM) approved and has been licensed by the United States Nuclear Regulatory Commission (NRC) for use in spent fuel storage technologies and spent fuel pools. The structural effect of the weight of the GT-Inserts on the fuel, the fuel rack, and the fuel pool structural interfaces and drop qualifications are unaffected. This is because the addition of five GT-Inserts (which increases the dry weight of a fuel assembly by 110 lbs.) brings the total weight to 1551 lbs. which is enveloped by the 2904 lbs. assumed in the calculation for fuel rack design.

Fuel Misloading

There is no significant increase in the probability of the accidental misloading of spent fuel assemblies into the spent fuel racks when assuming a complete loss of the Boraflex panels and considering the presence of soluble boron in the pool water for criticality control. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures and will be in accordance with Technical Specification (TS) 3.7.18[,] ''Spent Fuel Assembly Storage[,]'' and Licensee Controlled Specification (LCS) 4.0.100, ''Fuel Storage Patterns,'' which will specify spent fuel rack storage configuration limitations.

There is no increase in the consequences of the accidental misloading of a spent fuel assembly into the spent fuel racks. The criticality analysis, performed by SCE, demonstrates that the pool K_{eff} will be maintained less than or equal to 0.95 following an accidental misloading by the boron concentration of the pool. The proposed TS 3.7.17[,] ''Fuel Storage Pool Boron Concentration[,]'' will ensure that an adequate spent fuel pool boron concentration is maintained.

Change in Spent Fuel Temperature

There is no significant increase in the probability of either the loss of normal cooling to the spent fuel pool water or a decrease in pool water temperature from a large emergency makeup when assuming a complete loss of the Boraflex panels and considering the presence of soluble boron in the spent fuel pool water. A high proposed concentration (>2000 parts per million (ppm)) of soluble boron is consistent with current operating practices maintained in the spent fuel pool water. The proposed minimum boron concentration of 2000 ppm in TS 3.7.17 will ensure that an adequate concentration is maintained in the spent fuel pools.

A loss of normal cooling to the spent fuel pool water causes an increase in the temperature of the water passing through the stored fuel assemblies. This causes a decrease in the water density, and when coupled with the assumption of a complete loss of Boraflex, may result in a positive reactivity addition. However, the additional negative reactivity provided by the boron concentration limit in the proposed TS 3.7.17 will compensate for the increased reactivity which could result from a loss of spent fuel pool cooling. Because adequate soluble boron will be maintained in the spent fuel pool water to maintain K_{eff} less than or equal to 0.95, the consequences of a loss of normal cooling to the spent fuel pool will not be increased.

The thermal considerations of the fuel are unaffected by the presence of the GT-Inserts because the guide tube is designed for the presence of a CEA; therefore, it is not a primary coolant flow area. The fuel rack normal thermal cooling and malfunctioned blocked cooling scenarios are unaffected by the presence of the GT-Inserts in the fuel assemblies.

The proposed change does not involve an 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 consideration of criticality accidents in the spent fuel pool are not new or different. They have been analyzed in the Updated Final Safety Analysis Report (UFSAR) and in previous submittals to the NRC. Specific accidents considered and evaluated include fuel assembly drop, fuel assembly misloading in the racks, and spent fuel pool water temperature changes.

The possibility for creating a new or different kind of accident is not credible. Neither Boraflex [n]or soluble boron are accident initiators. The proposed change takes credit for soluble boron in the spent fuel pool while maintaining the necessary margin of safety. Because soluble boron has always been present in the spent fuel pool, a dilution of the spent fuel pool soluble boron has always been a possibility. However, a criticality accident resulting from a dilution accident was not considered credible. For this proposed amendment, SCE performed a spent fuel pool dilution analysis, which demonstrated that a dilution of the boron concentration in the spent fuel pool water which could increase the rack K_{eff} to greater than 0.95 (constituting a reduction of the required margin to criticality) is not a credible event. The requirement to maintain boron concentration in the spent fuel pool water for reactivity control will have no effect on normal pool operations and maintenance. There are no changes in equipment design or plant configuration.

The possibility of accidentally withdrawing a GT-Insert is minimized because special tooling is required to remove it, and it is completely contained within the guide tubes of the designated assemblies. Potential misloading of the GT-Inserts is minimized due to the design of the installation equipment, procedural controls, and double verification that will be in place to ensure the GT-Inserts are installed properly.

The possibility of accidentally withdrawing a CEA is minimized because specialized tooling is required for withdrawing a CEA from a fuel assembly. It is physically possible for the spent fuel handling tool to bind on a CEA after ungrappling from a fuel assembly and raising the tool. However, existing SONGS [San Onofre Nuclear Generating Station] procedures require that the operator validate ''tool weight only'' on the spent fuel handling machine's load cell read out after ungrappling from a fuel assembly and raising the hoist slightly, and to report this information to the engineer directing the fuel movement.

Therefore, the proposed change will not result in 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 TS changes proposed by this license amendment request and the resulting spent fuel storage operation limits will provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. Those limits are based on a San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 plant specific analysis that was performed in accordance with a methodology previously approved by the NRC.

The proposed change takes partial credit for soluble boron in the spent fuel pool. SCE's analyses show that spent fuel storage requirements meet the following NRC acceptance criteria for preventing criticality outside the reactor.

(1) The neutron multiplication factor, K_{eff}, including all uncertainties, shall be less than 1.0 when flooded with unborated water, and

(2) The neutron multiplication factor, K_{eff}, including all uncertainties, shall be less than or equal to 0.95 when flooded with borated water.

The criticality analysis utilized credit for soluble boron to ensure K_{eff} will be less than or equal to 0.95 under normal circumstances, and storage configurations have been defined using a 95/95 K_{eff} calculation to ensure that the spent fuel rack will be less than 1.0 with no soluble boron. Soluble boron credit is used to provide safety margin by maintaining K_{eff} less than or equal to 0.95 including uncertainties, tolerances[,] and accident conditions in the presence of spent fuel pool soluble boron. SCE evaluated the loss of a substantial amount of soluble boron from the spent fuel pool water which could lead to K_{eff} exceeding 0.95 and showed that it was not credible.

Also, the spent fuel rack K_{eff} will remain less than 1.0 with the spent fuel pool flooded with unborated water.

Decay heat, radiological effects, and seismic loads are unchanged by the absence of Boraflex.

The mechanical properties and the weight of the fuel assemblies remain essentially unchanged with the inclusion of the weight of five GT-Inserts per assembly. The original mechanical and thermal analysis of the fuel assembly/fuel rack and fuel pool building interfaces currently approved remain valid and conservative.

Therefore, the proposed change does not involve a significant reduction in the plant's 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: March 30, 2006.

Description of amendment request: The proposed amendments revise Technical Specification 3.3.3.6, ''Accident Monitoring Instrumentation,'' with respect to the required action for inoperable Wide Range Reactor Coolant Temperature, Wide Range Steam Generator Water Level, and Auxiliary Feedwater (AFW) Flow.

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 increase in the allowed outage times for the Reactor Coolant Outlet Temperature—Wide Range, Reactor Coolant Inlet Temperature—Wide Range, Steam Generator [Water] Level—Wide Range, and the AFW Flow does not involve a significant increase in the probability of an accident previously evaluated because these are accident monitoring functions that have no effect on the potential for accident initiation. The proposed deletion of the existing requirements in ACTION 38 is an administrative change. Since these requirements are not currently applied to any plant equipment, this change cannot affect the probability of any accident previously evaluated.

The proposed increase in the allowed outage times for the Reactor Coolant Outlet Temperature—Wide Range, Reactor Coolant Inlet Temperature—Wide Range, Steam Generator [Water] Level—Wide Range, and AFW Flow does not involve a significant increase in the consequences of an accident previously evaluated because the availability of redundant and diverse indications provides adequate assurance that the operator will be able to determine the post-accident status of the secondary heat sink.

The proposed deletion of the existing requirements in ACTION 38 is an administrative change. Since these requirements are not currently applied to any plant equipment, this change cannot affect the consequence 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 increase in the allowed outage times for the Reactor Coolant Outlet Temperature—Wide Range, Reactor Coolant Inlet Temperature—Wide Range, Steam Generator [Water] Level—Wide Range, and the AFW Flow does not create the possibility of a new or different kind accident from any accident previously evaluated because the proposed change affects only the allowed outage time for accident monitoring instrumentation and involves no changes to plant design, plant configuration or operating procedures.

The proposed deletion of the existing requirements in ACTION 38 is an administrative change. Since these requirements are not currently applied to any plant equipment, this change cannot create the possibility of any kind of accident.

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

Response: No.

The proposed increase in the allowed outage times for the Reactor Coolant Outlet Temperature—Wide Range, Reactor Coolant Inlet Temperature—

Wide Range, Steam Generator [Water] Level—Wide Range, and AFW Flow does not involve a significant reduction in the margin of safety because the availability of redundant and diverse indications provides adequate assurance that the operator will be able to determine the post-accident status of the secondary heat sink.

The proposed deletion of the existing requirements in ACTION 38 is an administrative change. Since these requirements are not currently applied to any plant equipment, this change cannot affect the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments 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, Units 1 and 2, Somervell County, Texas

Date of amendment request: February 21, 2006.

Brief description of amendments: The amendments revise Technical Specification (TS) 5.6.5 entitled, ''Core Operating Limits Report (COLR),'' to revise the listed Loss-of-Coolant Accident (LOCA) and non-LOCA analysis methodologies used at Comanche Peak Steam Electric Station, Units 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. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change involves an administrative change only. Designation of the accident analysis methodologies, described in ERX–04–004 and ERX–04–005, as approved analytical methods is required to maintain the accuracy of the Technical Specification 5.6.5 (Core Operating Limits Report) and to maintain consistency with the resolution of issues as prescribed in 10 CFR 50.46. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change involves an administrative change only. Technical Specification 5.6.5 is being changed to reference the revised accident analysis methodologies currently under NRC review. No actual plant equipment will be affected by the proposed change. 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.

Margin of safety is associated with the confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, Reactor Coolant System pressure boundary, and containment structure) to limit the level of radiation dose to the public. This request involves an administrative change (subject to NRC approval) only to incorporate the NRC-approved methodologies into the allowable analysis methodologies specified in Technical Specification 5.6.5. No actual plant equipment will be affected by the proposed change. The compliance of the revised methodology with the requirements of 10 CFR 50.46 and Appendix K will be addressed through the NRC staff's review of the topical reports. Therefore, it is concluded that the use of the proposed methodology will not degrade the confidence in the ability of the fission product barriers to limit the level of radiation dose to the public. 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.

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

Date of amendment request: February 21, 2006.

Brief description of amendments: The amendments would revise Technical Specifications (TS) 3.3.1, 3.3.2, 3.4.5, 3.4.6, and 3.4.7, ''Reactor Trip System (RTS) Instrumentation,'' ''Engineered Safety Feature System Actuation (ESFAS) Instrumentation,'' ''RCS [Reactor Coolant System] Mode 3,'' ''RCS Loops-Mode 4,'' and ''RCS Loops-Mode 5, Loops Filled,'' respectively. The revisions reflect the different steam generator water level trip setpoints and steam generator inventory requirements associated with the planned replacement of the steam generators in Comanche Peak Steam Electric Station, Unit 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. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed TS changes affect the protective and mitigative capabilities of the plant; none of the changes impact the initiation or probability of occurrence of any accident.

The consequences of accidents evaluated in the FSAR [Final Safety Analysis Report] that could be affected by this proposed change are those in which the steam generator water level trip functions are credited for initiating a protective or mitigative function. These transients and accidents have been analyzed and all relevant event acceptance criteria were shown to be satisfied. The radiological dose consequences are unaffected. Therefore, there is no increase in the consequences of an accident previously evaluated.

The actual proposed setpoint values were determined using an uncertainty methodology previously approved by the NRC for this application. These values provide adequate assurance that required protective and mitigative functions will be initiated as assumed in the transient and accident analyses. Therefore, there is no increase in the consequences of an accident previously evaluated.

The proposed revisions to the Δ76 steam generator inventory, required to ensure that the steam generators can provide an effective heat sink, are consistent with the current design requirements. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of

accident from any accident previously evaluated?

Response: No.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of these changes. 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 changes. No new equipment performance burdens are imposed. The possibility of a new or different malfunction of safety-related equipment is not created. 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 to the Steam Generator Water Level-Low-Low and Steam Generator Water Level-High-High trip function setpoints protect the assumed safety analysis limits established in the transient and accident analyses. When used in the transient and accident analyses, all relevant event acceptance criteria are satisfied. Therefore, these proposed changes do not result in the reduction in a margin of safety.

The proposed changes to the Δ76 steam generator inventory requirements, which ensure the steam generators can function as an effective heat sink during required shutdown operating modes, are consistent with the existing design and licensing bases. Therefore, these proposed changes do not result in the 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.

Georgia Power Company, Docket Nos. 50–321 and 50–366, Edwin I. Hatch Nuclear Plant, Unit Nos. 1 and 2, Appling County, Georgia

Date of amendment request: March 17, 2006.

Brief description of amendment request: The proposed amendment would add a license condition to Section 2.C of the Edwin I. Hatch Nuclear Plant, Unit Nos. 1 and 2, Operating Licenses. This license condition will authorize the licensee to credit administering potassium iodide (KI) to reduce the 30-day post-accident thyroid radiological dose to the operators in the main control room for an interim period of approximately 4 years. In addition, the design-basis accident analysis section of the Updated Final Safety Analysis Reports will be updated to reflect crediting of KI.

Date of publication of individual notice in ***Federal Register:*** March 27, 2006 (71 FR 15223).

Expiration date of individual notice: 30-day date April 26, 2006; 60-day date May 26, 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.*

Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: December 19, 2005.

Brief description of amendment: The amendment revised the Technical Specification (TS) to make permanent the temporary changes to TS Table 3.3.8.1–1 previously approved by Amendment No. 147. TS Table 3.3.8.1–1 is revised to delete the temporary note, correct the number of Required Channels per Division for the Loss of Power (LOP) time delay functions, and delete the requirement to perform Surveillance Requirement 3.3.8.1.2, the monthly Channel Functional Test, on certain LOP time delay functions.

Date of issuance: May 17, 2006.

Effective date: As of the date of issuance and shall be implemented prior to expiration of the temporary change on June 1, 2006, provided by Amendment No. 147.

Amendment No.: 151.

Facility Operating License No. NPF–47: The amendment revised the Technical Specfications.

Date of initial notice in ***Federal Register:*** March 14, 2006 (71 FR 13173).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 17, 2006.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Docket No. 50–278, Peach Bottom Atomic Power Station, Unit 3, York and Lancaster Counties, Pennsylvania

Date of application for amendment: July 6, 2005, as supplemented March 15 and April 7, 2006.

Brief description of amendments: The proposed changes extend the use of the Peach Bottom Atomic Power Station, Unit 3, pressure-temperature (P–T) limits specified in the Technical Specifications (TSs) from 22 to 32 effective full-power years.

Date of issuance: May 12, 2006.

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

Amendment No.: 263.

Renewed Facility Operating License No. DPR–56: The amendment revised the TSs.

Date of initial notice in ***Federal Register:*** August 2, 2005 (70 FR 44402). The supplements dated March 15, 2006, and April 7, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 12, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 21, 2005, as supplemented February 28, March 28 and April 24, 2006.

Brief description of amendment: The amendment revised the Operating License and Technical Specifications to allow operation of St. Lucie Unit 2 with a reduced reactor coolant system flow rate of 300,000 gpm and a reduction in the maximum thermal power to 89 percent of the rated thermal power. The flow rate of 300,000 gpm conservatively bounds an analyzed steam generator tube plugging level of 42 percent per steam generator.

Date of Issuance: May 16, 2006.

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

Amendment No.: 145.

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

Date of initial notice in ***Federal Register:*** December 20, 2005 (70 FR 75492). The February 28, March 28 and April 24, 2006, supplements did not affect the original proposed no significant hazards determination, or expand the scope of the request as noticed in the **Federal Register**.

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

No significant hazards consideration comments received: No.

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

Date of amendment request: September 22, 2005, as supplemented by letters dated March 24, 2006, and April 28, 2006.

Description of amendment request: The proposed amendment revised the Seabrook Station, Unit No. 1 Technical Specifications (TSS) to increase the licensed thermal power level by 1.7% to 3648 megawatts thermal.

Date of issuance: May 22, 2006.

Effective date: As of its date of issuance, and shall be implemented within 12 months.

Amendment No.: 110.

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

Date of initial notice in ***Federal Register:*** November 8, 2005 (70 FR 67748). The licensee's letters dated March 24, 2006, and April 28, 2006, provided clarifying information that did not change the scope of the proposed amendment as described in the original notice of proposed action published in the **Federal Register**, and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 22, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 29, 2005.

Brief description of amendments: The amendments revised Technical Specification 3.7.5, ''Auxiliary Feedwater (AFW) System,'' to change the frequency of Surveillance Requirement 3.7.5.6 from 92 days to 24 months.

Date of issuance: May 17, 2006.

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

Amendment Nos.: 186 and 188.

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

Date of initial notice in ***Federal Register:*** October 11, 2005 (70 FR 59086).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 17, 2006.

No significant hazards consideration comments received: No.

South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of application for amendment: November 29, 2005.

Brief description of amendment: This amendment for V. C. Summer revises TSs by eliminating the requirements to submit monthly operating reports and certain annual reports.

Date of issuance: May 19, 2006.

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

Amendment No.: 175.

Renewed Facility Operating License No. NPF–12: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 14, 2006 (71 FR 13178).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 19, 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 13, 2005.

Brief description of amendment: The amendment changes the steam generator (SG) level requirement for Limiting Condition for Operation 3.4.7.b and Surveillance Requirements 3.4.5.2, 3.4.6.3 and 3.4.7.2 from greater than or equal (≥) to 6 percent (%) to ≥ 32% following replacement of the SGs during the Unit 1, Cycle 7 refueling outage.

Date of issuance: May 5, 2006.

Effective date: As of the date of issuance and shall be implemented prior to entering Mode 5 upon restart from the Unit 1 Cycle 7 (U1C7) Refueling Outage.

Amendment No.: 61.

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

Date of initial notice in ***Federal Register:*** February 14, 2006 (71 FR 7814).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 8, 2005.

Brief description of amendments: These amendments revised the auxiliary feedwater (AFW) requirements of Technical Specifications (TSs) 3.6, ''Turbine Cycle,'' and 4.8, ''Auxiliary Feedwater System,'' to eliminate the inconsistency between the AFW pump requirements and the required actions, establish consistency with the Improved TSs, and add an AFW flowpath allowed outage time along with required actions.

Date of issuance: February 23, 2006.

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

Amendment Nos.: 246 and 245.

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

Date of initial notice in ***Federal Register:*** April 26, 2005 (70 FR 21465).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 23, 2006.

No significant hazards consideration comments received: No.

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

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

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

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

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

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

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

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. Within 60 days after the date of publication of this notice, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, and electronically on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If there are problems in accessing the document, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737, or by e-mail to *pdr@nrc.gov*. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

1. Technical—primarily concerns/issues relating to technical and/or health and safety matters discussed or referenced in the applications.
2. Environmental—primarily concerns/issues relating to matters discussed or referenced in the environmental analysis for the applications.
3. Miscellaneous—does not fall into one of the categories outlined above.

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

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

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

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

Southern California Edison Company, et al., Docket No. 50–362, San Onofre Nuclear Generating Station, Unit 3, San Diego County, California

Date of amendment request: May 4, 2006.

Description of amendment request: Allowed repairing a line in the shutdown cooling (SDC) system with the unit in Mode 4. This repair plan caused Unit 3 to be out of compliance with the licensing basis of the SDC system for the limited duration of the repair, but not to exceed 7 days.

Date of issuance: May 5, 2006.

Effective date: Immediate.

Amendment No.: 194.

Facility Operating License No. (NPF–15): Amendment revised the Updated Final Safety Analysis Report, Section 5.4.7.1.2.C. with a note that states that the change is only applicable from the date of issuance of the amendment until the repair is completed on the SDC line or 7 days, whichever occurs first.

Public comments requested as to proposed no significant hazards consideration (NSHC): No. The Commission's related evaluation of the amendment, finding of emergency circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated May 5, 2006.

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

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

NRC Branch Chief: David Terao.

Dated at Rockville, Maryland, this 25th day of May 2006.

For the Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. E6–8450 Filed 6–5–06; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

Regulatory Guide: Issuance, Availability

The U.S. Nuclear Regulatory Commission (NRC) has issued a new guide in the agency's Regulatory Guide Series. This series has been developed to describe and make available to the public such information as methods that are acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques that the staff uses in evaluating specific problems or postulated accidents, and data that the staff needs in its review of applications for permits and licenses.

Regulatory Guide 1.205, ''Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants,'' provides guidance for use in complying with the requirements that the NRC has promulgated for risk-informed, performance-based fire protection programs that meet the requirements of Title 10, § 50.48(c), of the *Code of Federal Regulations* (10 CFR 50.48(c)) and the referenced 2001 Edition of the National Fire Protection Association (NFPA) standard, NFPA 805, ''Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants.''

In accordance with 10 CFR 50.48(a), each operating nuclear power plant must have a fire protection plan that satisfies General Design Criterion (GDC) 3, ''Fire Protection,'' of Appendix A, ''General Design Criteria for Nuclear Power Plants,'' to 10 CFR part 50, ''Domestic Licensing of Production and Utilization Facilities.'' In addition, plants that were licensed to operate before January 1, 1979, must meet the requirements of 10 CFR part 50, Appendix R, ''Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979,'' except to the extent provided for in 10 CFR 50.48(b). Plants licensed to operate after January 1, 1979, are required to comply with 10 CFR 50.48(a), as well as any plant-specific fire protection license condition and technical specifications.

Section 50.48(c), which the NRC adopted in 2004 (69 FR 33536, June 16, 2004), incorporates NFPA 805 by reference, with certain exceptions, and allows licensees to voluntarily adopt and maintain a fire protection program that meets the requirements of NFPA 805 as an alternative to meeting the requirements of 10 CFR 50.48(b) or the plant-specific fire protection license conditions. Licensees who choose to comply with 10 CFR 50.48(c) must submit a license amendment application to the NRC, in accordance with 10 CFR 50.90. Section 50.48(c)(3) describes the required content of the application.

The Nuclear Energy Institute (NEI) has developed NEI 04–02, ''Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c),'' Revision 1, dated September 2005, to assist licensees in adopting 10 CFR 50.48(c) and making the transition from their current fire protection program (FPP) to one based on NFPA 805. This regulatory guide endorses NEI 04–02, Revision 1, because it provides methods acceptable to the NRC for implementing NFPA 805 and complying with 10 CFR 50.48(c), subject to the additional regulatory positions contained in Section C of this regulatory guide and the approval authority that 10 CFR 50.48(c) grants to the authority having jurisdiction (AHJ). The regulatory positions in Section C include clarification of the guidance provided in NEI 04–02, as well as any NRC exceptions to the guidance. The regulatory positions in Section C take precedence over the NEI 04–02 guidance.

All references to NEI 04–02 in this regulatory guide refer to Revision 1 of NEI 04–02. All references to NFPA 805 in this regulatory guide refer to the 2001 Edition of NFPA.

The NRC previously solicited public comment on this new guide by publishing a **Federal Register** notice (69 FR 60192) concerning Draft Regulatory Guide DG–1139 on October 7, 2004. Following the closure of the public comment period on December 15, 2004, the staff considered all stakeholder comments in the course of preparing Regulatory Guide 1.205. The NRC staff's responses to public comments received on the draft regulatory guide are available electronically in the NRC's Agencywide Documents Access and Management System (ADAMS) at *http://www.nrc.gov/reading-rm/adams.html*, under Accession #ML061100235. In particular, the revisions in this new guide include additional guidance regarding the plant change process, including risk acceptance thresholds for changes that may be made without prior NRC review and approval. In addition, this new guide includes guidance for the fire probabilistic safety analyses that licensees use to risk-inform the fire protection program.

The NRC staff encourages and welcomes comments and suggestions in connection with improvements to published regulatory guides, as well as items for inclusion in regulatory guides that are currently being developed. You may submit comments by any of the following methods.

Mail comments to: Rules and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

Hand-deliver comments to: Rules and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, 11555 Rockville Pike, Rockville, Maryland 20852, between 7:30 a.m. and 4:15 p.m. on Federal workdays.

Fax comments to: Rules and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, at (301) 415–5144.

Requests for technical information about Regulatory Guide 1.205 may be directed to Paul W. Lain at (301) 415–2346 or via e-mail to *PWL@nrc.gov*.

Regulatory guides are available for inspection or downloading through the NRC's public Web site in the Regulatory Guides document collection of the NRC's Electronic Reading Room at *http://www.nrc.gov/reading-rm/doc-collections*. Regulatory Guide 1.205 is also available electronically in the NRC's Agencywide Documents Access and Management System (ADAMS) at *http://www.nrc.gov/reading-rm/adams.html*, under Accession #ML061100174.

In addition, regulatory guides are available for inspection at the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville, Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555–0001. The PDR can also be reached by telephone at (301) 415–4737 or (800) 397–4205, by fax at (301) 415–3548, and by e-mail to *PDR@nrc.gov.* Requests for single copies of draft or final guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Reproduction and Distribution Services Section; by e-mail to *DISTRIBUTION@nrc.gov*; or by fax to