containment pressure boundary surface areas, a general visual-type examination, in accordance with the Hatch 1 and 2 Qualified (N) Coatings Program, is sufficient to inspect the subject surface areas of the containment and will provide an acceptable level of quality and safety.

In summary, the licensee is proposing an exemption from the requirements of Section 50.55a(b)(2)(ix)(G) to use an alternate examination method to examine Item E.20 of Table IWE–2500–1 of ASME Code, Section XI, pursuant to 10 CFR 50.12(a)(1) and 10 CFR 50.12(a)(2)(ii). The licensee stated in its application that compliance with the visual examination requirements of Section 50.55a(b)(2)(ix)(G) is not necessary for accessible surface areas of the containment vessel pressure retaining boundary Vent System to achieve the underlying purpose of the rule.

3.0 Discussion

Pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 50 when: (1) The exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and (2) when special circumstances are present. Special circumstances are present whenever, in accordance with 10 CFR Part 50.12(a)(2)(ii), ''Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule * * *.'' Therefore, in determining the acceptability of the licensee's exemption request, the NRC staff has performed the following evaluation to satisfy the requirements of 10 CFR 50.12 for granting the exemption.

The underlying purpose of 10 CFR 50.55a(b)(2)(ix)(G), as it applies to Item E1.20 of Table IWE–2500–1, is to ensure that an examination of the metal containment or the metal liner of a concrete containment is performed to identify corrosion or other degradation that could affect the structural or leak-tight integrity of the structure.

The NRC staff examined the licensee's rationale to support the exemption request and concluded that maintaining the integrity of the coating system applied to the Hatch 1 and 2 containment vent system components is a preventive measure that would protect against corrosion of the coated components. As the licensee emphasizes the effectiveness of its coating program, the NRC staff believes that the general visual examination performed as part of maintaining the integrity of the coating system is a proactive action and will ensure the integrity of the coated vent system components. The proposed alternative will provide the quality and safety level similar to the one intended by the use of VT–3 examination of the vent system components, and would meet the underlying purpose of 10 CFR Section 50.55a(b)(2)(ix)(G).

Based on a consideration of proposed alternatives contained in the licensee's letters dated March 20, and August 2 and 24, 2005, the NRC staff concludes that degradation of the containment structure would be detected using the proposed alternative, thus meeting the underlying purpose of the rule. Therefore, the NRC staff concludes that the proposed exemption from 10 CFR Section 50.55a(b)(2)(ix)(G) is acceptable.

4.0 Conclusion

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12, the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. Also, special circumstances are present. Therefore, the Commission hereby grants SNC an exemption from the requirement of 10 CFR Section 50.55a(b)(2)(ix)(G) to perform a VT–3 examination for Item E1.2 of Table IWE–2500–1, for Hatch 1 and 2, for the 4th 10-year ISI interval.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will not have a significant effect on the quality of the human environment (70 FR 76082).

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 6th day of January 2006.

For the Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. E6–415 Filed 1–13–06; 8:45 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 December 22, 2005 to January 5, 2006. The last biweekly notice was published on January 3, 2006 (71 FR 145).

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Date of amendment request: December 2, 2005.

Description of amendment request: The amendment would revise the Technical Specifications to increase the allowable as-found main steam safety valve code safety function lift setpoint tolerance from ±1% to ±3%.

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. Will operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed TS changes allow for an increase in the as-found Main Steam Safety Valve (MSSV) setpoint tolerance from ±1% to ±3%. The proposed changes do not alter the MSSV nominal lift setpoints or MSSV lift setpoint test frequency.

The proposed TS changes have been evaluated on both a generic and plant specific basis. The NRC has approved the general approach of this change; however, implementation is contingent on several plant specific evaluations. The required plant specific analyses and evaluations included transient analysis of the anticipated operational transients (AOTs); analysis of the design basis overpressurization event; evaluation of the performance of high pressure systems, and evaluation of the containment response during Loss-of-Coolant Accident (LOCA) and hydrodynamic loads on the MSSV discharge lines and containment. These analyses and evaluations demonstrate that there is adequate margin to the design core thermal limits and reactor vessel pressure limits using the ±3% MSSV as-found setpoint tolerance. The analyses and evaluations also demonstrate that the operation of high-pressure safety systems will not be adversely affected and that the containment response during a LOCA will be acceptable.

Evaluations of the impact of the proposed change on the equipment important to safety have been performed and no adverse conditions were identified. The reactor pressure vessel and attached systems and piping have been evaluated for the impact of this proposed TS change. A plant specific analysis has been performed which indicates that the ASME Code upset limits for the reactor pressure vessel will not be exceeded for the limiting event, i.e., Main Steam Isolation Valve (MSIV) closure with flux Scram. The reactor pressure vessel and attached piping design values will not be exceeded. Therefore, the probability of a malfunction of the reactor pressure vessel and attached systems and piping is not increased and the consequences of such an accident remain acceptable.

The nuclear fuel has been evaluated for the impact of the proposed change.

Plant specific analyses were performed which indicate that for all abnormal operational transients adequate margin to the fuel thermal limit parameters, i.e., Minimum Critical Power Ratio (MCPR) and thermal-mechanical limits, is maintained. Emergency Core Cooling System (ECCS)/LOCA performance is maintained adequate to meet the requirements of 10 CFR 50.46. Therefore, the consequences of these accidents remain acceptable and the probability of the malfunction of the nuclear fuel is not increased.

The Containment response during a LOCA has been evaluated for the impact of the proposed change. The major factor in the Containment pressure response to a LOCA is the rate of reactor vessel water inventory loss due to a DBA LOCA. The rate of reactor vessel water inventory loss is mainly dependent on the initial reactor pressure, which is not affected by the proposed setpoint tolerance change. The major factor in the Containment temperature response to a LOCA is the integrated steam inventory loss due to Main Steamline Break. The rate of reactor vessel steam inventory loss is mainly dependent on the reactor decay heat, which is not affected by the proposed setpoint tolerance change. Therefore, the consequences of these accidents remain acceptable and the probability of the malfunction of Containment is not increased.

The Control Rod Drive (CRD) system has been evaluated for the impact of the proposed change. The CRD system capability of controlling reactor power during normal plant operation and rapidly inserting control rod blades (Scram) during abnormal plant conditions is not impacted by the proposed change. Therefore, the probability of a malfunction of the CRD system is not increased.

The Reactor Vessel Instrumentation System has been evaluated for the impact of the proposed change. The Reactor Vessel Instrumentation System will continue to be operated within the current design pressure/temperature requirements; therefore, the probability of a malfunction of the Reactor Vessel Instrumentation System is not increased.

An administrative change is also being proposed to correct the reference to ''IWV–3510 of Section XI of the ASME Boiler and Pressure Vessel Code'' in TS 4.3.E because the stated ASME section no longer exists. The TS is being changed to reference specification 4.3.C for MSSV testing. This is an administrative change and does not affect previously evaluated accidents.

Therefore, the proposed TS changes do not significantly increase the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance of the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed TS changes allow for an increase in the as-found MSSV setpoint tolerance from ±1% to ±3%. Generic and plant specific analyses and evaluations indicate that the plant response to any previously evaluated event will remain acceptable. All plant systems, structures, and components will continue to be capable of performing their required safety function as required by event analysis guidance.

The proposed TS changes do not alter the MSSV nominal lift setpoints or MSSV lift setpoint test frequency. The operation and response of the affected equipment important to safety is unchanged. All systems, structures, and components will continue to be operated within acceptable operating and/or design parameters. No system, structure, or component will be subjected to a condition that has not been evaluated and determined to be acceptable using the guidance required for specific event analysis.

The change to correct the reference to ''IWV–3510 of Section XI of the ASME Boiler and Pressure Vessel Code'' in TS 4.3.E is an administrative change and does not affect the possibility of a new or different kind of accident.

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

3. Will operation of the facility in accordance with the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed TS changes allow for an increase in the as-found MSSV setpoint tolerance from ±1% to ±3%. The proposed TS changes do not alter the MSSV nominal lift setpoints or MSSV lift setpoint test frequency. The operation and response of the affected equipment important to safety is unchanged. All systems, structures, and components will continue to be operated within acceptable operating and/or design parameters. While the calculated peak reactor vessel pressure for the ASME overpressure event is higher than that calculated without the increase in setpoint tolerance, it is still within the respective licensing acceptance limits associated with this event. These licensing acceptance limits have been determined by the NRC to provide a sufficient margin of safety.

The increase in MSSV steam flow and reactor vessel pressure does not reduce the margin of safety associated with the MSSVs and associated components and structures since the increased MSSV steam flow rate and reactor vessel pressure are bounded by the current design analysis.

The margin of safety for fuel thermal limits and 10 CFR 50.46 limits are unaffected by the proposed change.

The margin of safety for the Containment is unaffected by the proposed change.

The capability of the SLC system and the CRD system to perform their safety functions during all required events, using the required guidance for event analysis, is maintained. Therefore, the proposed changes do not reduce the margin of safety provided by the SLC and CRD systems.

The change to correct the reference to ''IWV–3510 of Section XI of the ASME Boiler and Pressure Vessel Code'' in TS 4.3.E is an administrative change and does not affect the margin of safety.

Therefore, these proposed TS 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: Thomas S. O'Neill, Associate General Counsel, Exelon Generation Company, LCC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Darrell J. Roberts.

Calvert Cliffs Nuclear Power Plant, Inc., Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: November 3, 2005.

Description of amendments request: The proposed amendments would revise the accident source term in the design-basis radiological consequences analyses and the associated Technical Specifications (TSs), pursuant to section 50.67 of part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.67). The proposed amendments would provide for the full implementation of the alternate source term (AST) in accordance with the guidance in Regulatory Guide 1.183, ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.'' The proposed amendments would also increase the flow rate for the control room emergency ventilation system (CREVS) from 2000 to 10000 cubic feet per minute in TS 5.5.11, ''Ventilation Filter Testing Program,'' by means of a modification to the CREVS. In addition, automatic isolation dampers and radiation monitors will also be installed at access control heating, ventilating, and air conditioning (HVAC) unit no. RTU–1 and access control air conditioning unit no. 13.

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

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The results of the applicable radiological design basis accidents (DBAs) re-evaluation demonstrated that, with the requested changes, the dose consequences of these limiting events are within the regulatory limits and guidance provided by the Nuclear Regulatory Commission in 10 CFR 50.67 and Regulatory Guide 1.183 for AST methodology. The AST is an input to calculations used to evaluate the consequences of an accident and does not by itself affect the plant response or the actual pathway of the activity released from the fuel. It does, however, better represent the physical characteristics of the release such that appropriate mitigation techniques may be applied.

The change from the original source term to the new proposed AST is a change in the analysis method and assumptions and has no effect on accident initiators or causal factors that contribute to the probability of occurrence of previously analyzed accidents. Use of an AST to analyze the dose effect of DBAs shows that regulatory acceptance criteria for the new methodology continues to be met. Changing the analysis methodology does not change the sequence or progression of the accident scenario.

The proposed Technical Specification changes reflect the plant configuration that will either support implementation of the AST analyses or eliminate requirements that are no longer needed as a result of the revised DBA analyses. The equipment affected by the proposed changes is mitigative in nature and relied upon after an accident has been initiated. The operation of various filtration systems have been considered in the evaluations for these proposed changes. While the operation of some systems does change with the implementation of an AST, the affected systems are not accident initiators; and application of the AST methodology, itself, is not an initiator of a DBA.

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

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

As described in Item 1 above, the changes proposed in this license amendment request involve the use of a new analysis methodology and related regulatory acceptance criteria. The proposed Technical Specification changes reflect the plant configuration that will either support implementation of the new methodology or eliminate requirements that are no longer needed as a result of the new methodology. No new or different accidents result from utilizing the proposed changes. Although the proposed changes require modification to the Control Room emergency ventilation system and installation of automatic isolation dampers and radiation monitors at Access Control HVAC Unit RTU–1 and Access Control Air Conditioning Unit 13 on the Auxiliary Building roof, none of these changes can initiate a new or different kind of accident since they are only related to system capabilities that provide protection from accidents that have already occurred. As a result, no new failure modes are being introduced that could lead to different accidents. These changes do not alter the nature of events postulated in the Updated Final Safety Analysis Report nor do they introduce any unique precursor mechanisms.

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

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

As described in Item 1 above, the changes proposed in this license amendment request involve the use of a new analysis methodology and related regulatory acceptance criteria. The proposed Technical Specification changes reflect the plant configuration that will either support implementation of the new methodology or eliminate requirements that are no longer needed as a result of the new methodology. Safety margins and analytical conservatisms have been evaluated and have been found acceptable. The analyzed events have been carefully selected and, with plant modification, margin has been retained to ensure that the analyses adequately bound postulated event scenarios. The analyses have been performed using conservative methodologies, as specified in Regulatory Guide 1.183. The dose consequences of these DBAs remain within the acceptance criteria presented in 10 CFR 50.67, ''Accident Source Term,'' and Regulatory Guide 1.183. The proposed changes continue to ensure that the doses at the exclusion area boundary and low population zone boundary, as well as the Control Room, are within corresponding regulatory limits.

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 amendments request involves no significant hazards consideration.

Attorney for licensee: Carey Fleming, Sr. Counsel—Nuclear Generation, Constellation Generation Group, LLC, 750 East Pratt Street, 17th floor, Baltimore, MD 21202.

NRC Branch Chief: Richard J. Laufer.

Exelon Generation Company, LLC, Docket No. 50–352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of amendment request: December 14, 2005.

Description of amendment request: The proposed amendment modifies the Technical Specifications (TSs) to incorporate a revised Single Loop Operation Safety Limit Minimum Critical Power Ratio (SLO SLMCPR) due to the cycle-specific analysis.

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 derivation of the cycle specific Single Loop Operation Safety Limit Minimum Critical Power Ratio (SLO SLMCPR) for incorporation into the Technical Specifications (TS), and its use to determine cycle-specific thermal limits, has been performed using the methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–15 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–15–US, September, 2005, which includes Amendment 25. Amendment 25 was approved by the NRC in a March 11, 1999 safety evaluation report.

The basis of the SLO SLMCPR calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The new SLO SLMCPR preserves the existing margin to transition boiling. The GE–14 fuel is in compliance with Amendment 22 to ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–15 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–15–US, September 2005, which provides the fuel licensing acceptance criteria. The probability of fuel damage will not be increased as a result of this change. Therefore, the proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The SLO SLMCPR is a TS numerical value, calculated to ensure that transition boiling does not occur in 99.9% of all fuel rods in the core if the limit is not violated. The new SLO SLMCPR is calculated using NRC approved methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–15 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–15–US, September 2005, which includes Amendment 25. Additionally, the GE–14 fuel is in compliance with Amendment 22 to ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–15 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–15–US, September, 2005, which provides the fuel licensing acceptance criteria. The SLO SLMCPR is not an accident initiator, and its revision will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

There is no significant reduction in the margin of safety previously approved by the NRC as a result of the proposed change to the SLO SLMCPR, which includes the use of GE–14 fuel. The new SLO SLMCPR is calculated using methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–15 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–15–US, September, 2005, which includes Amendment 25. The SLO SLMCPR ensures that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated when all uncertainties are considered, thereby preserving the fuel cladding integrity.

Therefore, the proposed TS change will not involve a significant reduction in [a] margin of safety previously approved by the NRC.

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

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

NRC Branch Chief: Darrell J. Roberts.

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

Date of amendment request: December 21, 2005.

Description of amendment request: The proposed amendment revises the Technical Specifications by relocating the Pressure Isolation Valve (PIV) tables to the Technical Requirements Manual (TRM).

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

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

Response: No.

The proposed relocation of Technical Specification Table 3.4.3.2–1 does not alter the requirements for pressure isolation valve operability or surveillance currently in the Technical Specifications. The proposed change to remove the pressure isolation valve table from TS and relocate the information to an administratively controlled document, and to revise the wording in TS to reflect this change, will have no impact on any safety related structures, systems or components. The probability of occurrence of a previously evaluated accident is not increased because this change does not introduce any new potential accident initiating conditions. The consequences of accidents previously evaluated in the UFSAR [Updated Final Safety Analysis Report] are not affected because the ability of the PIVs to limit leakage through these valves in amounts that do not compromise safety is not affected. 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 accident previously evaluated?

Response: No.

The proposed changes are administrative in nature and do not result in physical alterations or changes in the method by which any safety related system performs its intended function(s). The proposed changes do not impact any safety analysis assumptions. The proposed changes do not create any new accident initiators or involve an activity that could be an initiator of an accident of a different type.

All PIVs and alarm instrumentation will continue to be tested to the same rigorous requirements as defined in the Technical Specification Surveillance Requirements. The proposed revision does not make changes in any method of testing or how any safety related system performs its safety functions. Therefore, the possibility of an accident of a different type than any previously evaluated in the UFSAR is not created.

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

Response: No.

The administrative change to relocate Technical Specification Table 3.4.3.2–1 to the Technical Requirements Manual does not alter the basic regulatory requirement for Reactor Coolant System pressure isolation and will not affect the isolation capability for credible accident scenarios. Future revisions to the Technical Requirements Manual Table will be subject to evaluation pursuant to 10 CFR 50.59.

Additionally, the proposed relocation does not alter the requirements for pressure isolation valve and alarm instrumentation operability currently in the Technical Specifications. The LCO [limiting condition for operation] and Surveillance Requirements will be retained in the revised Technical Specifications. The proposed change will not affect the meaning, application, and function of the current Technical Specification requirements for the valves in Table 3.4.3.2–1. Therefore, the proposed changes do not result in a significant reduction in [a] margin of safety.

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

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

NRC Branch Chief: Darrell J. Roberts.

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

Date of amendment request: December 15, 2005.

Description of amendment request: The proposed change revises the values of the safety limit minimum critical power ratio (SLMCPR) in Technical Specification (TS) section 2.1.1, ''Reactor Core SLs.'' Specifically, the proposed change would require that for Unit 2, the minimum critical power ratio (MCPR) for Global Nuclear Fuel (GNF) fuel shall be ≥1.09 for two recirculation loop operation, or ≥1.10 for single recirculation loop operation. Additionally, the proposed change would require that MCPR for Westinghouse fuel shall be ≥1.11 for two recirculation loop operation, or ≥1.13 for single recirculation loop operation.

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:

According to 10 CFR 50.92, ''Issuance of amendment,'' paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if 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.

EGC has evaluated the proposed change to the TS for QCNPS, Unit 2, using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

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

Response: No.

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits have been established consistent with NRC-approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. The proposed change conservatively establishes the SLMCPR for QCNPS, Unit 2, Cycle 19 such that the fuel is protected during normal operation and during plant transients or anticipated operational occurrences (AOOs).

Changing the SLMCPR does not increase the probability of an evaluated accident. The change does not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. Therefore, no individual precursors of an accident are affected.

The proposed change revises the SLMCPR to protect the fuel during normal operation as well as during plant transients or AOOs. Operational limits will be established based on the proposed SLMCPR to ensure that the SLMCPR is not violated. This will ensure that the fuel design safety criterion (i.e., that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and AOOs) is met. Since the proposed change does not affect operability of plant systems designed to mitigate any consequences of accidents, the consequences of an accident previously evaluated are not expected to increase.

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.

Creation of the possibility of a new or different kind of accident would require creating one or more new accident precursors. New accident precursors may be created by modifications of plant configuration, including changes in allowable modes of operation.

The proposed change does not involve any plant configuration modifications or changes to allowable modes of operation. The proposed change to the SLMCPR assures that safety criteria are maintained for QCNPS, Unit 2, Cycle 19.

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

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

Response: No.

The SLMCPR provides a margin of safety by ensuring that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and AOOs if the MCPR limit is not violated. The proposed change will ensure the appropriate level of fuel protection by continuing to ensure that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and AOOs if the MCPR limit is not violated. Additionally, operational limits will be established based on the proposed SLMCPR to ensure that the SLMCPR is not violated. This will ensure that the fuel design safety criteria (i.e., that no more than 0.1% of the rods are expected to be in boiling transition if the MCPR limit is not violated) are met.

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

Based upon the above, EGC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

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

Attorney for licensee: Mr. Brad Fewell, Assistant General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Acting Branch Chief: Mindy S. Landau.

First Energy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1 (PNPP), Lake County, Ohio

Date of amendment request: November 21, 2005.

Description of amendment request: The proposed amendment would revise the acceptance criteria of Technical Specification (TS) Surveillance Requirements (SRs) associated with TS 3.8.1, ''AC Sources—Operating,'' to modify the Emergency Diesel Generator (EDG) start tests to provide minimum voltage and frequency limits and clarify other limits as steady state parameters. Specifically, the amendment would revise SRs 3.8.1.2, 3.8.1.7, 3.8.1.12, 3.8.1.15 and 3.8.1.20. This change is consistent with the approved Technical Specification Task Force Traveler (TSTF) 163, Revision 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. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change is a LAR (license amendment request) that modifies the acceptance criteria for the PNPP TS SRs pertaining to the EDGs. The EDGs mitigate the consequences of previously evaluated accidents involving a loss of offsite power. The EDGs are used to support mitigation of the consequences of an accident, but they are not considered as the initiator of any previously analyzed accident.

The proposed LAR does not change the manner in which the EDGs are operated and when implemented will continue to ensure the EDGs perform their function when called upon. The proposed revision to the TS SRs will continue to ensure that minimum frequency and voltage are attained within the required time. The SRs will continue to ensure that proper steady state voltage and frequency are attained consistent with proper EDG governor and voltage regulator performance.

The proposed LAR does not affect the design of the EDGs, the operational characteristics of the EDGs, the interfaces between the EDGs and other plant systems, the function, or reliability of the EDGs. Thus, the EDGs will be capable of performing their accident mitigation function and there is no impact to the radiological consequences of any accident analysis.

As such, the proposed change continues to provide adequate assurance of operable EDGs and does not involve any increase to the probability or consequences of an accident previously evaluated.

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

The proposed LAR introduces no new mode of plant operation and it does not involve physical modification to the plant. New equipment is not installed with the proposed LAR, nor does the proposed LAR cause existing equipment to be operated in a new or different manner.

Since the proposed changes do not involve a change to the plant design or operation, no new system interactions are created by this change. The proposed LAR does not produce any parameters or conditions that could contribute to the initiation of accidents different from those already evaluated in the Updated Safety Analysis Report.

The changes to the affected TS SRs do not affect the assumed accident performance of the EDGs, nor any plant structure, system or component previously evaluated.

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

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

The proposed change is a LAR that does not impact EDG performance, including the capability for each EDG to attain and maintain required voltage and frequency for accepting and supporting plant safety loads within the required time, as assumed in the plant safety analysis.

The proposed LAR does not involve a significant reduction in a margin of safety since the operability of the EDGs continues to be determined as required to support the capability of the EDGs to provide emergency power to plant equipment that mitigate the consequences of an accident.

The proposed LAR does not introduce changes to setpoints or limits established or assumed by the accident analysis. Therefore, implementation of the proposed LAR 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: David W. Jenkins, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Branch Chief: Mindy Landau, Acting.

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

Date of amendment request: December 6, 2005.

Description of amendment request: The proposed amendment would revise the Seabrook Station, Unit No. 1 Technical Specification 3.8.3.1, ''Onsite Power Distribution,'' to extend the allowed outage time for balance-of-plant vital inverters 1–EDE–I–1E and 1–EDE–I–1F from 24 hours to 7 days.

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

The proposed change extends the allowed outage time (AOT) for the balance-of-plant (BOP) instrument bus inverters from 24 hours to 7 days. The BOP instrument bus inverters do not solely support any risk-significant functions. The failure of an inverter is not an initiator of any analyzed event and does not increase the frequency of an initiating event. Consequently, extending the AOT will not have an impact on the frequency of occurrence of any event previously analyzed. The proposed change does not alter the design, configuration, operation, or function of any plant system, structure, or component. As a result, the outcomes of previously evaluated accidents are unaffected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The proposed change does not challenge the performance or integrity of any safety-related system. The proposed change neither installs nor removes any plant equipment, not alters the design, physical configuration, or mode of operation of any plant structure, system, or component. Installed equipment will not be operated in a new or different manner. No physical changes are being made to the plant, so no new accident causal mechanisms are being introduced. Procedures that ensure the unit operates within analyzed limits and procedures that respond to off-normal and emergency conditions are not altered with this proposed change. Therefore, the proposed change does not create the possibility of a new or different accident from any previously evaluated.

3. The proposed changes do not involve a significant reduction in [a] margin of safety.

The margin of safety associated with the acceptance criteria of any accident is unchanged. The proposed change does not alter the design, configuration, operation, or function of any plant system, structure, or component. The ability of any operable structure, system, or component to perform its designated safety function is unaffected by this change. Operation with one instrument bus inverter inoperable and the associated instrument bus aligned to its maintenance supply does not result in a significant reduction in [a] margin of safety. Surveillance testing of the emergency diesel generators (EDGs) and the electrical distribution system provides confidence that the EDGs will energize the emergency AC buses following a loss of power. 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: M. S. Ross, Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408–0420.

NRC Branch Chief: Darrell J. Roberts.

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

Date of amendment request: November 12, 2004.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 5.5.7, ''Inservice Testing Program,'' and TS 5.5.8, ''Steam Generator (SG) Tube Surveillance Program,'' to update references to the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code) and certain associated periodicities for inservice testing activities consistent with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) section 50.55a, ''Codes and standards.''

The proposed amendment would also correct a typographical error contained in TS 5.5.8.b.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. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.

The proposed change revises Technical Specifications for consistency with the requirements of 10 CFR 50.55a(f)(4) and 10 CFR 50.55a(g)(4).

The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves.

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

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

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

The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing. The proposed change does not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure.

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

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

The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing. The safety function of the affected components will be maintained.

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

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

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

NRC Branch Chief: L. Raghavan.

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

Date of amendment request: October 11, 2005.

Description of amendment request: The proposed amendment would revise certain 18-month Technical Specification (TS) Surveillance Requirements (SRs) to eliminate the condition that testing be conducted during shutdown.

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

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

Response: No.

The proposed changes permit PSEG to evaluate the conditions required to safely perform a TS SR. These surveillance tests verify that equipment will perform its intended safety function of mitigating an accident. No analyzed accident scenario is being revised. The initiating conditions and assumptions for accidents described in the Hope Creek Generating Station Updated Final Safety Analysis Report (UFSAR) remain as previously analyzed.

The proposed changes do not reduce the ability of the mitigating equipment to perform its safety function. The TS will continue to require the surveillance tests to be performed on an eighteen-month periodicity to verify operability. As a result, the ability of the mitigating equipment to perform its safety function is unaffected by the proposed change.

The capitalization change is proposed to improve readability and does not alter any requirement.

Based upon the above, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

Response: No.

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated in the UFSAR. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. Specifically, no new hardware is being added to the plant as part of the proposed change, no existing equipment is being modified, and no significant changes in operations are being introduced.

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 will not alter any assumptions, initial conditions, or results of any accident analyses. The proposed changes to remove the requirement to perform certain testing during shutdown conditions allows PSEG to evaluate the conditions needed to safely perform the required testing. There is no change to the frequency of testing or in the testing that is required. There is no change in the responsibility of PSEG to perform tests in a safe and responsible manner. Any changes to procedures will have to be individually evaluated to ensure that they do not reduce the margin of safety. The changes do not affect the ability of systems, structures or components to perform their safety related functions. In addition, the proposed changes do not affect the ability of the safety systems to ensure that the facility can be maintained in a shutdown or refueling condition for extended periods of time.

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

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

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

NRC Branch Chief: Darrell J. Roberts.

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

Date of amendment request: August 31, 2005; as supplemented December 8, 2005.

Description of amendment request: The proposed amendment would relocate the containment high range accident monitors from the radiation monitoring instrumentation technical specification (TS) to the accident monitoring TS and correct a typographical error contained in a previous amendment.

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 presents no change in the probability of a previously evaluated accident.

The proposed change presents no change in the consequence of an accident, since the containment high range accident monitors are used post-accident to determine the amount of core damage and status of the fission product barriers.

The containment high range accident monitors are used post accident to assess the conditions inside containment. They have an automatic function to switch the subcooling margin monitor (SCMM) to ''adverse'' mode (i.e., it displays a more conservative indication of the amount of subcooling in the RCS) [reactor coolant system]. Additionally, the containment high range accident monitors provide an indication that is used post accident in determining the status of the fission product barriers. There will be no change in the operation or use of the containment high range accident monitors.

The remaining change is editorial in nature and does not impact the accident analysis in any manner.

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

Response: No.

The proposed change is a minor change that is administrative in nature. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. No new hardware is added, existing hardware is not modified and no significant changes in operations are implemented. Post accident monitoring instrumentation is not associated with the initiation of an accident.

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

Response: No.

The proposed change does not alter the manner in which safety limits, limiting safety systems settings or limiting conditions for operation are determined. The proposed change will not alter any assumptions, initial conditions or results specified in any accident analysis.

There is no change in the containment high range accident monitor high level alarm setpoint. The ECS [electronic check source] is functionally equivalent to the TS definition of SOURCE CHECK.

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

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

NRC Branch Chief: Darrell J. Roberts.

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

Date of amendment request: September 21, 2005.

Description of amendment request: The amendment would change the scope of steam generator (SG) tube inspections required in the SG tubesheet 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.

Of the various accidents previously evaluated, the proposed changes only affect the steam generator tube rupture (SGTR) event evaluation and the postulated steam line break (SLB) accident evaluation. Loss-of-coolant accident (LOCA) conditions cause a compressive axial load to act on the tube. Therefore, since the LOCA tends to force the tube into the tubesheet rather than pull it out, it is not a factor in this amendment request. Another faulted load consideration is a safe shutdown earthquake (SSE); however, the seismic analysis of Westinghouse 51 Series SGs has shown that axial loading of the tubes is negligible during an SSE.

PSEG's amendment request takes credit for how the tubesheet enhances the tube integrity in the Westinghouse Electric Company explosive tube expansion (WEXTEX) region by precluding tube deformation beyond its initial expanded outside diameter. For the SGTR and SLB events, the required structural margins of the SG tubes will be maintained due to the presence of the tubesheet. Tube rupture is precluded for axial cracks in the WEXTEX region due to the constraint provided by the tubesheet. Therefore, the normal operating 3ΔP margin and the postulated accident 1.43ΔP margin against burst are maintained.

The W* length supplies the necessary resistive force to preclude pullout loads under both normal operating and accident conditions. The contact pressure results from the WEXTEX expansion process, thermal expansion mismatch between the tube and tubesheet, and from the differential pressure between the primary and secondary side. Therefore, the proposed change results in no significant increase in the probability or the occurrence of an SGTR or SLB accident.

The proposed changes do not affect other systems, structures, components or operational features. Therefore, based on the above evaluation, the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

The consequences of an SGTR event are primarily affected by the primary-to-secondary flow rate and the time duration of the primary-to-secondary flow during the event. Primary-to-secondary flow rate through a postulated ruptured tube (i.e., complete severance of a single SG tube) is not affected by the proposed change since the flow rate is based on the inside diameter of a[n] SG tube and the pressure differential. PSEG's amendment request does not change either of these. The duration of primary-to-secondary leakage is based on the time required for an operator to determine that a[n] SGTR has occurred, the time to identify and isolate the faulted SG, and ensure termination of radioactive release to the atmosphere from the faulted SG. PSEG's amendment request does not affect the duration of the primary-to-secondary leakage because it does not change the control room indicators with which an operator would determine that an SGTR has occurred. The consequences of an SGTR are secondarily affected by primary-to-secondary leakage, which could occur due to axial cracks remaining in service in the WEXTEX region in a non-faulted SG. During a[n] SGTR, the primary-to-secondary differential pressure is less than or equal to the normal operating differential pressure; therefore, the primary-to-secondary leakage due to axial cracks in the WEXTEX region of a non-faulted SG during a[n] SGTR would be less than or equal to the primary-to-secondary leakage experienced during normal operation. Primary-to-secondary leakage is considered in the calculation determining the consequences of a[n] SGTR and the value is bounding.

The postulated SLB has the greatest primary-to-secondary pressure differential, and therefore could experience the greatest primary-to-secondary leakage. PSEG's amendment request requires the aggregate leakage, (i.e., the combined leakage for the tubes with service induced degradation inside the tubesheet) to remain below the maximum allowable SLB primary-to-secondary leakage rate limit such that the doses are maintained to less than the 10 CFR [Part] 100 limits and also less than the GDC-[General Design Criterion]19 limits.

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.

PSEG's amendment request does not introduce any physical changes to the Salem Unit 2 SGs. PSEG's amendment request takes credit for how the tubesheet enhances the SG tube integrity in the WEXTEX region. Because degradation detected within the W* distance are required to be plugged, it is highly unlikely that a tube would fail as a result of a circumferential defect. Therefore a tube severance, which would strike neighboring tubes and create a multiple tube rupture, is not credible. The proposed change does not introduce any new equipment or any change to existing equipment. No new effects on existing equipment are created. Based on the above evaluation, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The amendment request maintains the structural margins of the SG tubes for both normal and accident conditions that are required by Regulatory Guide 1.121. For cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. WCAP–14797, Revision 2 defines a length W* of degradation free expanded tubing, that provides the necessary resistance to tube pullout due to the pressure induced forces (with applicable safety factor applied). Application of the W* methodology will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for large margins between calculated and actual leakage values in the W* criteria.

Based on the above, it is concluded that the proposed changes do not result in a significant reduction of margin with respect to plant safety as defined in the Updated Final Analysis Report or Technical Specifications. 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: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Branch Chief: Darrell J. Roberts.

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

Date of amendment request: December 1, 2005.

Description of amendment request: The proposed amendment would change the SSES–1 Technical Specifications (TSs) by revising the Unit 1 Cycle 15 (U1C15) minimum critical power ratio (MCPR) safety limit for single loop operation in section 2.1.1.2 and references listed in TS 5.6.5.b.

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

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

Response: No.

The proposed change to the single-loop MCPR Safety Limit does not directly or indirectly affect any plant system, equipment, component, or change the processes used to operate the plant. Further, the proposed U1C15 MCPR Safety Limit was generated using NRC approved methodology and meets the applicable acceptance criteria. Thus, this proposed amendment does not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated.

Prior to the startup of U1C15, licensing analyses are performed (using NRC approved methodology referenced in Technical Specification Section 5.6.5.b) to determine changes in the critical power ratio as a result of anticipated operational occurrences. These results are added to the MCPR Safety Limit values to generate the MCPR operating limits in the U1C15 COLR [core operating limits report]. These limits could be different from those specified for the current Unit 1 COLR. The COLR operating limits thus assure that the MCPR Safety Limit will not be exceeded during normal operation or anticipated operational occurrences. Postulated accidents are also analyzed prior to the startup of U1C15 and the results shown to be within the NRC approved criteria.

The changes to the references in Section 5.6.5.b were made to properly reflect the NRC approved methodology used to generate the U1C15 core operating limits. The use of this approved methodology does not increase the probability of occurrence or consequences of an accident previously evaluated.

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

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

Response: No.

The change to the single-loop MCPR Safety Limit does not directly or indirectly affect any plant system, equipment, or component and therefore does not affect the failure modes of any of these items. Thus, the proposed change does not create the possibility of a previously unevaluated operator error or a new single failure. The changes to the references in Section 5.6.5.b were made to properly reflect the NRC approved methodology used to generate the U1C15 core operating limits. The use of this approved methodology does not create the possibility of a new or different kind of accident.

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

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

Response: No.

Since the proposed changes do not alter any plant system, equipment, component, or the processes used to operate the plant, the proposed change will not jeopardize or degrade the function or operation of any plant system or component governed by Technical Specifications. The proposed single-loop MCPR Safety Limit does not involve a significant reduction in the margin of safety as currently defined in the Bases of the applicable Technical Specification sections, because the MCPR Safety Limits calculated for U1C15 preserve the required margin of safety.

The changes to the references in section 5.6.5.b were made to properly reflect the NRC approved methodology used to generate the U1C15 core operating limits. This approved methodology is used to demonstrate that all applicable criteria are met, thus, demonstrating that there is no reduction in 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: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Branch Chief: Richard J. Laufer.

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

Date of amendment request: October 5, 2005.

Description of amendment request: The proposed amendment would revise the SSES 1 and 2 Technical Specifications (TSs) 3.4.10, ''RCS [reactor coolant system] Pressure and Temperature (P/T) Limits,'' to remove valid P/T curve limit date and replacing it with the effective full-power years (EFPY) of radiation exposure on each of the P/T limit curves for SSES 1 and 2. The new P/T limit would be 35.7 EFPY for SSES 1 and 30.2 EFPY for SSES 2.

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

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

No. The proposed changes request that the P/T limits curves in TS 3.4.10, ''RCS Pressure and Temperature (P/T) Limits'' be revised by removing the valid date and replacing it with the Effective Full Power Years of radiation exposure limit on each of the P/T curves for SSES Units 1 and 2.

The P/T limits are prescribed during all operational conditions to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate, resulting in nonductile failure of the reactor coolant pressure boundary, an unanalyzed condition. Therefore, the proposed changes do not have any effect on the probability of an accident previously evaluated.

The P/T curves are used as operational limits during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region. The P/T curves provide assurance that station operation is consistent with previously evaluated accidents. Thus, the radiological consequences of an accident previously evaluated are not increased.

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 accident previously evaluated?

No. The proposed changes do not change the response of any plant equipment to transient conditions. The proposed changes do not introduce any new equipment, modes of system operation, or failure mechanisms.

Therefore, there are no new types of failures or new or different kinds of accidents or transients that could be created by these changes. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The consequences of a previously evaluated accident are not increased by these proposed changes, since the Loss of Coolant Accident analyzed in the FSAR [Final Safety Analysis Report] assumes a complete break of the reactor coolant pressure boundary. The changes to the P/T limits curves do not change this assumption.

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: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Branch Chief: Richard J. Laufer.

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

Date of amendment request: November 9, 2004, as supplemented December 15, 2005. This notice supersedes the original notice published on April 26, 2005 (70 FR 21463), which was based upon the licensee's application dated November 9, 2004.

Description of amendment request: The proposed amendments would change the SSES 1 and 2 Technical Specifications (TSs) 3.8.4, ''DC Sources— Operating,'' 3.8.5, ''DC Sources—Shutdown,'' 3.8.6, ''Battery Cell Parameters,'' and add a new TS section, 5.5.13, ''Battery Monitoring and Maintenance Program.'' These changes are consistent with Technical Specification Change Traveler (TSTF) 360, Revision 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. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed changes restructure the Technical Specifications (TSs) for the DC Electrical Power Systems. The proposed changes consist of the relocation of several surveillance requirements that perform preventive maintenance on the safety related batteries, to a new license controlled program. The DC electrical power systems, including associated battery chargers, are not initiators to any accident sequence analyzed in the Final Safety Analysis Report (FSAR). Operation in accordance with the proposed TS ensures that the DC electrical power systems are capable of performing functions as described in the FSAR. Therefore, the mitigative functions supported by the DC Power Systems will continue to provide the protection assumed by the analysis.

The relocation of preventive maintenance surveillance, and certain operating limits and actions to a newly created, licensee-controlled TS 5.5.13, ''Battery Monitoring and Maintenance Program,'' will not challenge the ability of the DC electrical power systems to perform their design functions. The maintenance and monitoring required by current TS, which are based on industry standards, will continue to be performed. In addition, the DC Power Systems are within the scope of 10 CFR 50.65, ''Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,'' which will ensure the control of maintenance activities associated with the DC electrical power systems. The integrity of fission product barriers, plant configuration, and operating procedures as described in the FSAR will not be affected by the proposed changes.

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 accident previously evaluated?

No. The proposed changes involve restructuring the TS for the DC electrical power systems. These changes will rely on a new license controlled program to monitor battery parameters for operability. The DC electrical power systems, which include the associated battery chargers, are not initiators to any accident sequence analyzed in the FSAR. Rather, the DC electrical power systems are used to supply equipment used to mitigate an accident. These mitigative functions, supported by the DC electrical power systems are not affected by these changes and they will continue to provide the protection assumed by the safety analysis described in the FSAR. There are no new types of failures or new or different kinds of accidents or transients that could be created by these changes.

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

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

No. The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed changes will not adversely affect operation of plant equipment. These changes will not result in a change to the setpoints at which protective actions are initiated. Sufficient DC electrical system capacity is ensured to support operation of mitigation equipment. The changes associated with the new Battery Maintenance and Monitoring Program will ensure that the station batteries are maintained in a highly reliable state. The equipment fed by the DC electrical sources will continue to provide adequate power to safety related loads in accordance with analysis assumptions.

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: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

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

Date of application request: October 26, 2005.

Description of amendment request: The amendment would revise Technical Specification (TS) 3.6.6, ''Containment Spray and Cooling Systems,'' to change Required Action D.1 that currently allows 72 hours of operation with both containment cooling trains out of service as long as both containment spray trains are operable. The required action would be revised to impose the more stringent requirement of requiring plant shutdown if both containment cooling trains are out of service instead of allowing the 72 hours to restore an inoperable train. There are also changes to other required actions in TS 3.6.6 to reflect the revision to Required Action D.1. In addition, the required action for two inoperable containment spray trains is being revised.

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 in the required action when two containment cooling trains are inoperable to require plant shutdown is more restrictive than the current required action that allows 72 hours of operation [to restore one containment cooling train to operable status]. Also the proposed change to the required action [F.1 for] when two containment cooling trains are inoperable to be in MODE 3 within 6 hours and MODE 5 within 36 hours [are the same as in the current Required Actions E.1 and E.2 for when the two containment cooling trains are inoperable. The proposed change to the required action for two containment spray trains being inoperable] is more restrictive than the current required action to enter LCO [Limiting Condition for Operation] 3.0.3 immediately [because] LCO 3.0.3 requires the plant to be in MODE 3 within 7 hours. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistently with the safety analysis and licensing basis [for Callaway].

All of these proposed changes have been reviewed to ensure no previously evaluated accident has been adversely affected. [The proposed changes are not accident initiators.] 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 does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in controlling [plant] parameters. The proposed change does impose different requirements. However, these changes are consistent with [the] assumptions made in the safety analysis and licensing basis [for Callaway]. 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 imposition of more stringent requirements has no impact on or will increase the margin of safety. The change in the required action when two containment cooling trains are out of service will increase the margin of safety by decreasing the allowed restoration time [to restore an inoperable containment cooling train to operable status].

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

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

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

NRC Branch Chief: David Terao.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: December 17, 2004.

Brief description of amendment: The amendment revised Appendix B, Environmental Technical Specifications, of the OCNGS Facility Operating License, principally by deleting redundant reporting requirements, aligning various requirements with regulations and accepted guidance documents, and correcting administrative errors.

Date of Issuance: January 4, 2006.

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

Amendment No.: 257.

Facility Operating License No. DPR–16: The amendment revised the Environmental Technical Specifications.

Date of initial notice in ***Federal Register:*** April 12, 2005 (70 FR 19113).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 25, 2005, as supplemented by letter dated August 4, 2005.

Brief description of amendment: The amendment revised the Millstone Power Station, Unit No. 2, Technical Specifications Surveillance Requirement for trisodium phosphate to remove the granularity term and chemical detail. In addition, the proposed change will increase the allowed outage time from 48 to 72 hours.

Date of issuance: January 3, 2006.

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

Amendment No.: 290.

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

Date of initial notice in ***Federal Register:*** July 19, 2005 (70 FR 41444). The additional information provided in the supplemental letter dated August 4, 2005, did not expand the scope of the application as noticed and did not change the NRC staff's original proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 16, 2004, as supplemented on October 5, 2005.

Brief description of amendment: The amendment revised the current fuel rod average licensing basis burnup limit for one lead test assembly containing advanced zirconium based alloys to a limit not exceeding 71,000 megawatt-days per metric ton of uranium.

Date of issuance: December 30, 2005.

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

Amendment No.: 228

Facility Operating License No. NPF–49: The amendment revised the design basis.

Date of initial notice in ***Federal Register:*** February 1, 2005 (70 FR 5238). The October 5, 2005, supplement provided clarifying information 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 December 30, 2005.

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: July 20, 2005, as supplemented by letter dated September 14, 2005.

Brief description of amendment: The amendment approves the transfer of Facility Operating License and Materials License No. NPF–38, held by Entergy Louisiana, Inc. (ELI) and Entergy Operatings, Inc. (EOI), for the Waterford Steam Electric Station, Unit 3 (Waterford 3). The transfer is associated with the restructuring of ELI from a Louisiana corporation to a Texas limited liability company, Entergy Louisiana, LLC (ELL). EOI will continue to operate Waterford 3, and the restructuring will not affect the technical or financial qualifications of ELL or EOI.

Date of issuance: December 31, 2005.

Effective date: At the time the transfer is completed.

Amendment No.: 203.

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

Date of initial notice in ***Federal Register:*** October 17, 2005 (70 FR 60374).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 2, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 17, 2004.

Brief description of amendments: The amendments revised the Appendix B, Environmental Technical Specifications.

Date of issuance: January 3, 2006.

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

Amendments Nos.: 257 and 260.

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

Date of initial notice in ***Federal Register:*** April 12, 2005 (70 FR 19112).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 3, 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: March 28, 2005, as supplemented September 23, 2005.

Description of amendment request: The amendment extended the expiration of Facility Operating License (FOL) NPF–86 for Seabrook Station, Unit No. 1, by approximately 3.4 years. The extension sets the date of expiration of the FOL to occur 40 years from the date of issuance of the full-power operating license. Specifically, the FOL, with a previous expiration date of October 17, 2026, now expires March 15, 2030. This change allows the recapture of zero-power and low-power testing time in accordance with SECY–98–296, "Agency Policy Regarding Licensee Recapture of Low-Power Testing or Shutdown Time for Nuclear Power Plants," dated December 21, 1998.

Date of issuance: December 28, 2005.

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

Amendment No.: 105.

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

Date of initial notice in ***Federal Register:*** May 24, 2005 (70 FR 29797). The licensee's September 23, 2005 supplement 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 December 28, 2005.

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: March 31, 2005, as supplemented November 9, 2005.

Brief description of amendment: This amendment extended the date for the next Appendix J, Type A test at St. Lucie Unit 2 until the end of the SL2–17 refueling outage.

Date of Issuance: December 23, 2005.

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

Amendment No.: 140.

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

*Date of initial notice in **Federal Register:*** June 7, 2005 (70 FR 33215). The November 9, 2005, supplement 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 December 23, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 29, 2004, as supplemented by letters dated May 6 and October 31, 2005.

Brief description of amendments: The amendments revised the Technical Specification (TS) requirements for the handling of irradiated fuel in the containment and fuel building, and certain specifications related to performing core alterations. These changes are based on analysis of the postulated fuel handling and core alteration accidents and transients for Diablo Canyon Nuclear Power Plant, Units 1 and 2. The amendments are consistent with the NRC-approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specifications Change Traveler, TSTF–51, Revision 2, ''Revise containment requirements during handling irradiated fuel and core alterations.'' In addition, the amendments made editorial corrections to TS 3.1.7, ''Rod Position Indication,'' TS 3.3.1, ''Reactor Trip System (RTS) Instrumentation,'' TS 3.4.16, ''RCS Specific Activity,'' TS 3.7.3, ''Main Feedwater Isolation Valve (MFIVs), Main Feedwater Regulating Valves (MFRVs), MFRV Bypass Valves, and Main Feedwater Pump (MFWP) Turbine Stop Valves,'' and TS 3.7.13, ''Fuel Handling Building Ventilation System (FHBVS).''

Date of issuance: January 3, 2006.

Effective date: January 3, 2006, and shall be implemented within 90 days of issuance.

Amendment Nos.: Unit 1—184; Unit 2—86.

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

*Date of initial notice in **Federal Register:*** January 4, 2005 (70 FR 403)

The supplements dated May 6 and October 31, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 3, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 12, 2004, as supplemented by letters dated September 2 and September 16, 2005.

Brief description of amendments: The amendments revised the Technical Specifications (TS) 3.1.7, ''Standby Liquid Control (SLC) System,'' for Hatch, Units 1 and 2. The amendments update Figure 3.1.7–1 and 3.1.7–2 of the Units 1 and 2 TS to reflect the increased concentration of Boron-10 in the solution. Conforming revisions to Bases B3.1.7, are also included.

Date of issuance: January 5, 2006.

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

Amendment Nos.: 247/191.

Renewed Facility Operating License Nos. DPR–57 and NPF–5: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 1, 2005 (70 FR 5249).

The supplemental letter dated September 2, 2005, contained clarifying information only and did not change the initial proposed no significant hazards consideration determination or expand the scope of the original **Federal Register** notice. The supplemental letter dated September 16, 2005, contained information that expanded the scope of the original **Federal Register** notice. The proposed amendment was re-noticed on October 25, 2005 (70 FR 61662).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 5, 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: April 27, 2005, as supplemented by letter dated November 17, 2005.

Brief description of amendments: The amendments relocate several Technical Specification (TS) requirements to the Sequoyah Technical Requirements Manual (TRM). Specifically, the amendments relocate the provisions for TS 3.3.2 (Movable Incore Detectors), TS 3.3.3.4 (Meteorological Instrumentation), TS 3.4.7 (Reactor Coolant System Chemistry), TS 3.4.11 (Reactor Coolant System Head Vents), TS 3.7.2 (Steam Generator Pressure and Temperature Limitations), TS 3.7.10 (Sealed Source Contamination), TS 3.9.5 (Refueling Operations Communications), and TS 3.9.6 (Manipulator Crane) to the TRM. These changes are consistent with the latest version of NUREG–1431, Revision 3, ''Standard Technical Specifications for Westinghouse Plants,'' and do not diminish the level of safety found in the current TSs.

Date of issuance: December 28, 2005.

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

Amendment Nos.: 305, 295.

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

*Date of initial notice in **Federal Register:*** July 5, 2005 (70 FR 38723). The supplemental letter of November 17, 2005, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 28, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 17, 2004.

Brief Description of amendments: These amendments revised the reactor coolant pressure and temperature limits, low-temperature overpressure protection system (LTOPS) setpoint values, and LTOPS enable temperatures that are valid for up to 47.6 effective full-power years (EFPY) and 48.1 EFPY of operation at Surry Power Station, Unit Nos. 1 and 2, respectively.

Date of issuance: January 3, 2006.

Effective date: As of the date of issuance and shall be implemented within 180 days from the date of issuance.

Amendment Nos.: 245/244.

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

Date of initial notice in ***Federal Register:*** March 1, 2005 (70 FR 9999).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 3, 2006.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 9th day of January 2006.

For the Nuclear Regulatory Commission.

Edwin M. Hackett,

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

[FR Doc. 06–320 Filed 1–13–06; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF MANAGEMENT AND BUDGET

Proposed Risk Assessment Bulletin

AGENCY: Office of Management and Budget.

ACTION: Notice of proposed Bulletin and request for comments.

SUMMARY: As part of an ongoing effort to improve the quality, objectivity, utility, and integrity of information disseminated by the Federal Government to the public, the Office of Management and Budget (OMB), in consultation with the Office of Science and Technology Policy (OSTP), has referred to the National Academy of Sciences (NAS), for their expert review, new guidance to enhance the quality and objectivity of risk assessments produced by the Federal Government. OMB will also be accepting public comment on this document until June 15, 2006.

DATES: Written comments regarding OMB's Proposed Risk Assessment Bulletin are due by June 15, 2006. This date has been selected in order to permit the public to participate in a related workshop to be organized by the NAS, prior to submitting their written comments.

ADDRESSES: Because of potential delays in OMB's receipt and processing of mail, respondents are strongly encouraged to submit comments electronically to ensure timely receipt. We cannot guarantee that comments mailed will be received before the comment closing date. Electronic comments may be submitted to: *OMB_RAbulletin@omb.eop.gov.* Please put the full body of your comments in the text of the electronic message and as an attachment. Please include your name, title, organization, postal address, telephone number and e-mail address in the text of the message. Please be aware that all comments are available for public inspection. Accordingly, please do not submit comments containing trade secrets, confidential or proprietary commercial or financial information, or other information that you do not want to be made available to the public. Comments also may be submitted via facsimile to (202) 395–7245.

FOR FURTHER INFORMATION CONTACT: Dr. Nancy Beck, Office of Information and Regulatory Affairs, Office of Management and Budget, 725 17th Street, NW., New Executive Office Building, Room 10201, Washington, DC 20503. Telephone (202) 395–3093.

SUPPLEMENTARY INFORMATION: OMB is seeking comments on its Proposed Risk Assessment Bulletin by June 15, 2006. The proposed Risk Assessment Bulletin is posted on OMB's Web site, *http://www.whitehouse.gov/omb/inforeg/infopoltech.html#iq.*

John D. Graham,

Administrator, Office of Information and Regulatory Affairs.

[FR Doc. E6–345 Filed 1–13–06; 8:45 am]

BILLING CODE 3110–01–P

PENSION BENEFIT GUARANTY CORPORATION

Exemption From the Bond/Escrow Requirement Relating to the Sale of Assets by an Employer Who Contributes to a Multiemployer Plan; LA Team Co. LLC

AGENCY: Pension Benefit Guaranty Corporation.

ACTION: Notice of exemption.

SUMMARY: The Pension Benefit Guaranty Corporation has granted a request from the LA Team Co. LLC for an exemption from the bond/escrow requirement of section 4204(a)(1)(B) of the Employee Retirement Income Security Act of 1974, as amended, with respect to the Major League Baseball Players Pension Plan. A notice of the request for exemption from the requirement was published on July 7, 2005 (70 FR 39349). The effect of this notice is to advise the public of the decision on the exemption request.

ADDRESSES: The non-confidential portions of the request for an exemption and the PBGC response to the request may be obtained by writing PBGC's Communications and Public Affairs Department (''CPAD'') at Suite 1200, 1200 K Street, NW., Washington, DC 20005–4026, or by visiting or calling CPAD (202–326–4040) during normal business hours.

FOR FURTHER INFORMATION CONTACT: Gennice D. Brickhouse, Office of the Chief Counsel, Suite 340, 1200 K Street, NW., Washington, DC 20005–4026; telephone 202–326–4020. (For TTY/TDD users, call the Federal Relay Service toll-free at 1–800–877–8339 and ask to be connected to 202–326–4020).

SUPPLEMENTARY INFORMATION:

Background

Section 4204 of the Employee Retirement Income Security Act of 1974, as amended by the Multiemployer Pension Plan Amendments Act of 1980 (''ERISA'' or ''the Act''), provides that a bona fide arm's-length sale of assets of a contributing employer to an unrelated party will not be considered a withdrawal if three conditions are met. These conditions, enumerated in section 4204(a)(1)(A)–(C), are that:

(A) The purchaser has an obligation to contribute to the plan with respect to the operations for substantially the same number of contribution base units for which the seller was obligated to contribute;

(B) The purchaser obtains a bond or places an amount in escrow, for a period of five plan years after the sale, in an amount equal to the greater of the seller's average required annual contribution to the plan for the three plan years preceding the year in which the sale occurred or the seller's required annual contribution for the plan year preceding the year in which the sale occurred (the amount of the bond or escrow is doubled if the plan is in reorganization in the year in which the sale occurred); and

(C) The contract of sale provides that if the purchaser withdraws from the plan within the first five plan years