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Dated: May 31, 2007.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 07–2802 Filed 6–1–07; 11:43 pm]

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NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from May 11, 2007, to May 23, 2007. The last biweekly notice was published on May 22, 2007 (72 FR 28717).

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, Rulemaking, Directives and Editing 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.*

Entergy Operations, Inc., Docket Nos. 50–313 and 50–368, Arkansas Nuclear One, Units 1 and 2 (ANO–1 and ANO–2), Pope County, Arkansas

Date of amendment request: April 24, 2007.

Description of amendment request: The proposed amendment will delete the Fuel Handling Area Ventilation System (FHAVS) and associated Ventilation Filter Testing Program (VFTP) requirements that are included in the ANO–1 Technical Specifications (TSs) 3.7.12 and 5.5.11 and the ANO–2 TSs 3.9.11 and 6.5.11. These requirements will be relocated to a licensee-controlled document, the unit-specific Technical Requirements Manuals (TRM), which are controlled under 10 CFR 50.59, ''Changes, tests, and experiments.''

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

1. [Do] the proposed change[s] involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The FHAVS is not involved in the initiation of any accidents. The system maintains a suitable environment for equipment operation and personnel access. They are also designed to filter any gaseous radioactivity that may occur during normal or accident conditions (i.e., a fuel handling accident). On this basis, the system is currently classified and designed as an Engineered Safety Features (ESF) air cleanup system. The FHAVS is used during movement of irradiated fuel, crane operation with loads over the Spent Fuel Pool (SFP), fuel shipments, and spent resin transfer to pull possible airborne radioactivity from the Train Bay by re-positioning manual dampers.

Revised ANO–1 and ANO–2 analysis of the dose consequences of a[n] FHA, to both the public and to the control room operator, demonstrate that doses remain well within regulatory acceptance limits without crediting filtration.

Thus there is no required safety function for the ANO–1 or ANO–2 FHAVS.

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

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

Response: No.

The FHAVS is not involved in the initiation of any accidents. It was designed to filter any gaseous radioactivity that may occur during normal or accident conditions (i.e., a fuel handling accident). No physical modifications are planned to the ANO–1 or ANO–2 FHAVS.

Revised ANO–1 and ANO–2 analysis of the dose consequences of a[n] FHA, to both the public and to the control room operator, demonstrate that doses remain well within regulatory acceptance limits without crediting filtration. Thus, there is no required safety function for the ANO–1 or ANO–2 FHAVS.

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

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

Response: No.

The FHAVS was designed to filter any gaseous radioactivity that may occur during normal or accident conditions (i.e., a fuel handling accident). No physical modifications are planned to the ANO–1 or ANO–2 FHAVS.

Revised ANO–1 and ANO–2 analysis of the dose consequences of a[n] FHA, to both the public and to the control room operator, demonstrate that doses remain well within regulatory acceptance limits without crediting filtration. The margin of safety, as defined in Standard Review Plan 15.7.4, Revision 1, and GDC [General Design Criterion] 19 has not been significantly reduced.

Therefore, the proposed change[s] [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: Terence A. Burke, Associate General Council—Nuclear Entergy Services, Inc., 1340 Echelon Parkway, Jackson, Mississippi 39213.

NRC Branch Chief: Thomas G. Hiltz.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: April 24, 2007.

Description of amendment request: The proposed amendment will revise Arkansas Nuclear One, Unit 2 (ANO–2) Technical Specification (TS) 5.2.1, ''Fuel Assemblies,'' to add Optimized ZIRLO™ as an acceptable fuel rod cladding material.

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 NRC approved topical report WCAP–12610–P–A and CENPD–404–P–A, Addendum 1–A ''Optimized ZIRLO™,'' prepared by Westinghouse Electric Company, LLC (Westinghouse), addresses Optimized ZIRLO™ and demonstrates that Optimized ZIRLO™ has essentially the same properties as currently licensed ZIRLO™. The fuel cladding itself is not an accident initiator and does not affect accident probability. Use of Optimized ZIRLO™ fuel cladding has been shown to meet all 10 CFR 50.46 design criteria and, therefore, will not increase the consequences of an accident.

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

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

Response: No.

Use of Optimized ZIRLO™ clad fuel will not result in changes in the operation or configuration of the facility. Topical report WCAP–12610–P–A and CENPD–404–P–A demonstrated that the material properties of Optimized ZIRLO™ are similar to those of standard ZIRLO™. Therefore, Optimized ZIRLO™ fuel rod cladding will perform similarly to those fabricated from standard ZIRLO™, thus precluding the possibility of the fuel becoming an accident initiator and causing a new or different type of accident.

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

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

Response: No.

The proposed change will not involve a significant reduction in the margin of safety because it has been demonstrated that the material properties of the Optimized ZIRLO™ are not significantly different from those of standard ZIRLO™. Optimized ZIRLO™ is expected to perform similarly to standard ZIRLO™ for all normal operating and accident scenarios, including both loss-of-coolant accident (LOCA) and non-LOCA scenarios. For LOCA scenarios, where the slight difference in Optimized ZIRLO™ material properties relative to standard ZIRLO™ could have some impact on the overall accident scenario, plant-specific LOCA analyses using Optimized ZIRLO™ properties will be performed prior to the use of fuel assemblies with fuel rods containing Optimized ZIRLO™. These LOCA analyses will demonstrate that the acceptance criteria of 10 CFR 50.46 will be satisfied when Optimized ZIRLO™ fuel rod cladding is implemented.

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: Terence A. Burke, Associate General Council—Nuclear Entergy Services, Inc., 1340 Echelon Parkway, Jackson, Mississippi 39213.

NRC Branch Chief: Thomas G. Hiltz.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: May 8, 2007.

Description of amendment request: The proposed amendment will revise Arkansas Nuclear One, Unit 2 (ANO–2) Technical Specification (TS) 3.1.1.4, ''Moderator Temperature Coefficient (MTC),'' to change the surveillance frequency to be based on effective full-power days instead of boron concentration.

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 continues to perform the SRs [surveillance requirements] to determine MTC at test intervals associated with the beginning and middle of the cycle. The results of the test[s] will continue to verify that the predicted MTC is consistent with the measured [MTC].

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

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

Response: No.

The proposed change does not result in any plant modifications or changes in the way the plant is operated. The revised SRs for confirming the MTC predicted values will continue to be performed at intervals associated with the beginning and middle of the cycle.

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

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

Response: No.

The proposed change does not result in any changes to the test method or to the frequency of the test. The change of the test interval to use EFPD [effective full-power days] instead of RCS [reactor coolant system] boron concentration still provides assurance that the predicted MTC is consistent with the measured [MTC].

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: Terence A. Burke, Associate General Council—Nuclear Entergy Services, Inc., 1340 Echelon Parkway, Jackson, Mississippi 39213.

NRC Branch Chief: Thomas G. Hiltz.

Exelon Generation Company, LLC, Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Units 1 and 2, Will County, Illinois

Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois.

Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois.

Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois.

Docket No. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania.

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

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.

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois.

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

Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania.

Date of amendment request: April 12, 2007.

Description of amendment request: The proposed amendment would modify technical specification (TS) requirements related to control room envelope (CRE) habitability in accordance with Technical Specification Task Force (TSTF) Traveler TSTF–448, Revision 3, ''Control Room Habitability.''

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 does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change revises the TS for the CRE emergency ventilation system, which is a mitigation system designed to minimize unfiltered air leakage into the CRE and to filter the CRE atmosphere to protect the CRE occupants in the event of accidents previously analyzed. An important part of the CRE emergency ventilation system is the CRE boundary. The CRE emergency ventilation system is not an initiator or precursor to any accident previously evaluated. Therefore, the probability of any accident previously evaluated is not increased. Performing tests to verify the operability of the CRE boundary and implementing a program to assess and maintain CRE habitability ensure that the CRE emergency ventilation system is capable of adequately mitigating radiological consequences to CRE occupants during accident conditions, and that the CRE emergency ventilation system will perform as assumed in the consequence analyses of design basis accidents. Thus, the consequences of any accident previously evaluated are not increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not impact the accident analysis. The proposed change does not alter the required mitigation capability of the CRE emergency ventilation system, or its functioning during accident conditions as assumed in the licensing basis analyses of design basis accident radiological consequences to CRE occupants. No new or different accidents result from performing the new surveillance or following the new program. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a significant change in the methods governing normal plant operation. The proposed change does not alter any safety analysis assumptions and is consistent with current plant operating practice. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The proposed change does not affect safety analysis acceptance criteria. The proposed change will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without compensatory measures. The proposed change does not adversely affect systems that respond to safely shut down the plant and to maintain the plant in a safe shutdown condition. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Russell Gibbs.

Exelon Generation Company, LLC, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois

Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Units 1 and 2, Will County, Illinois.

Date of amendment request: April 4, 2007.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 5.5.16, ''Containment Leakage Rate Testing Program,'' to reflect a one-time deferral of the containment Type A, integrated leak rate test from once in 10 years to once in 15 years.

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 changes will revise Braidwood Station and Byron Station TS 5.5.16, ''Containment Leakage Rate Testing Program'' to reflect a one-time, five-year extension of the containment Type A test date to enable the implementation of a 15-year test interval.

The containment is designed to contain radioactive material that may be released from the reactor core following a design basis

Loss of Coolant Accident (LOCA). The test interval associated with Type A testing is not a precursor of any accident previously evaluated. Type A testing does provide assurance that the containment will not exceed allowable leakage rate criteria specified in the TS and will continue to perform its design function following an accident. A risk assessment of the proposed changes has concluded that there is an insignificant increase in total population dose rate and an insignificant increase in the conditional containment failure probability.

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

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

The proposed changes for a one-time, five-year extension of the Type A tests for Braidwood Station and Byron Station will not affect the control parameters governing unit operation or the response of plant equipment to transient and accident conditions. The proposed changes do not introduce any new equipment, modes of system operation or failure mechanisms.

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

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

The Braidwood Station and Byron Station containment consists of the concrete containment building, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a design basis LOCA. Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. The cylinder wall is pre-stressed with a post[-] tensioning system in the vertical and horizontal directions, and the dome roof is pre-stressed utilizing a three way post-tensioning system.

The concrete containment building is required for structural integrity of the containment under Design Basis Accident (DBA) conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment.

The integrity of the containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak tight integrity of the containment is verified by a Type A integrated leak rate test (ILRT) as required by 10 CFR 50, Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.'' These tests are performed to verify the essentially leak tight characteristics of the containment at the design basis accident pressure.

The existing 10-year Type A test interval is based on past performance. Previous Type A leakage tests conducted at Braidwood Station Units 1 and 2, and Byron Station Units 1 and 2 indicate that leakage from containment has been less than the 10 CFR 50 Appendix J leakage limit.

The proposed changes for a one-time extension of the Type A tests do not affect the method for Type A, B or C testing or the test acceptance criteria. Type B and C testing will continue to be performed at the frequency required by the Braidwood Station and Byron Station Technical Specifications. The containment inspections that are performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI and 10 CFR 50.65, ''Requirements for monitoring the effectiveness of maintenance at nuclear power plants,'' provide a high degree of assurance that the containment will not degrade in a manner that is only detectable by Type A testing.

In NUREG–1493, ''Performance-Based Containment Leak-Test Program,'' the NRC indicated that a 20-year extension for Type A testing resulted in an imperceptible increase in risk to the public. The NUREG–1493 study also concluded that, generically, the design containment leak rate contributes a very small amount to the individual risk [and] have a minimal affect on this risk. EGC has conducted risk assessments to determine the impact of a change to the Braidwood Station and Byron Station Type A test schedule from a baseline value of once in 10 years to once in 15 years for the risk measures of Large Early Release Frequency (LERF), Total Population Dose, and Conditional Containment Failure Probability (CCFP). The results of the risk assessments indicate that the proposed changes to the Braidwood Station and Byron Station Type A test schedule has a minimal impact on public risk.

Therefore, based on previous Type A test results for the Braidwood Station and Byron Station containments, the current containment surveillance programs at each station, and the results of the EGC risk assessments, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Russell Gibbs.

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

Date of amendment request: August 7, 2006, as supplemented by letters dated January 22, and May 14, 2007, which included a revised no significant hazards consideration determination (NSHCD). This NSHCD is from the May 14, 2007, supplement.

Description of amendment request: The proposed amendment would revise the Seabrook Station Unit No. 1 (Seabrook) Facility Operating License (FOL) and Technical Specifications (TSs). The proposed changes would correct a joint-owner name in the operating license, remove a license condition from Appendix C to the FOL that is no longer applicable, and remove the list of Bases sections from the TS Index. Additionally, the proposed amendment would remove two manual valves from TS table 3.3.9, ''Remote Shutdown System,'' and add the requirement that only one charging pump is permitted to be aligned for injection into the reactor coolant system (RCS) in Modes 4, 5, and 6 to TS 3.4.9.3, ''Overpressure Protection Systems.'' The additional requirement proposed for TS 3.4.9.3 would allow for two pumps to be aligned for injection under administrative controls for up to one hour to permit swap over operations. The proposed changes would also remove a 1-hour reporting requirement for portable makeup pump system storage from TS 3.7.4, ''Service Water System/Ultimate Heat Sink,'' correct an error in TS 4.7.4.3, related to the service water pumphouse water level and delete a footnote from TS 3.7.6.2, ''Air Conditioning,'' that was only applicable to Cycle 7. The proposed changes would also delete a redundant reporting requirement in TS 6.6, ''Safety Limit Violation.'' Lastly, the proposed amendment would modify TS 6.7.6, ''Radioactive Effluent Controls Program,'' to clarify the TS with respect to the performance of dose projections.

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 probability or consequences of accidents previously evaluated in the UFSAR [Updated Final Safety Analysis Report] are unaffected by this proposed change. There is no change to any equipment response or accident mitigation scenario, and this change results in no additional challenges to fission product barrier integrity. 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.

This change limits availability of the charging pumps to one pump when in Mode 4 with the temperature of any RCS cold leg is less than or equal to 290 °F, in Mode 5, and in Mode 6 with the reactor vessel head on and the vessel head closure bolts not fully de-tensioned. Nonetheless, imposing this limitation does not alter the configuration or operation of the charging pumps from that specified in current administrative controls. Technical Specification (TS) 3/4.5.3, ECCS [Emergency Core Cooling System] Subsystems—Tavg Less Than 350 °F, presently stipulates that only one charging pump is maintained operable in Mode 4. Similarly, Technical Requirement 26, Boration Systems, requires that all but one operable charging pump be demonstrated inoperable in Modes 4, 5, and 6. Also, the Seabrook Station Updated Final Safety Analysis Report (UFSAR) describes the configuration of the charging pumps during shutdown conditions: Prior to decreasing RCS temperature below 350 °F, the safety injection pumps and the non-operating charging pumps are made inoperable. Consequently, the change does not alter the configuration or operation of the charging pumps from the procedures presently described in the UFSAR; rather, it only relocates an existing limitation from the UFSAR to the technical specifications. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

This proposed change also revises the minimum water level in the service water system pump house required for operability of the service water system. The value currently specified in the technical specifications has been in error since 1986 and will be corrected with this change. Increasing the minimum required water level from five feet to 25.1 feet does not alter the configuration or operation of the service water system. Following discovery of this discrepancy, administrative controls established a minimum water level of approximately 25 feet. Moreover, monitoring of the service water pump house level during 2005 observed that the level, which is controlled by the ocean tides, is normally greater than 26 feet. During this period the minimum and maximum pump house water levels were 26.3 and 48.57 feet, respectively. This administrative change has no affect on the actual operation or configuration of the service water system. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed revision to TS Table 3.3–9, Remote Shutdown System, eliminates valves MS–V127 and MS–V128 from the table. Located in the main steam supply line to the turbine-driven emergency feedwater (TDEFW) pump, these are locked open, manually operated, valves. Supplement 4 of NUREG 0896, Safety Evaluation Report, discusses the modifications made to the Emergency Feedwater System (EFW) to address problems experienced with the EFW steam supply lines during hot functional testing. A design change, installed in 1991, changed MS–V127 and MS–V128 to normally open valves, replaced the valves' pneumatic actuators with gear-operated manual operators, and re-assigned the EFW actuation and containment isolation functions of these valves to new automatic isolation valves (MS–V393 and MS–V394) in the TDEFW pump steam supply line. As a result, the elimination of MS–V127 and MS–V128 from TS Table 3.3–9 does not alter the design, configuration, operation, or function of these valves with regard to operation of the EFW system because in the existing design these normally open valves are not required to re-position to support operation of the TDEFW pump. Automatic valves MS–V393 and MS–V394, which actuate to initiate operation of the TDEFW pump, are appropriately under the control of TS Table 3.3–9. This proposed change does not alter the design, configuration, operation, or function of the EFW steam supply valves. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The other changes in this proposed amendment correct errors, remove an outdated license condition, remove an inconsistency between indexes, and revise a reporting requirement. These changes are administrative in nature and do not impact the design, configuration, operation, or function of any plant system, structure, or component. Therefore, the proposed changes do 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 previously evaluated.

The proposed changes (1) relocate an existing limitation from the UFSAR to the technical specifications regarding availability of the charging pumps, (2) revise the minimum water level in the service water system pump house required for operability of the service water system, (3) eliminate valves MS–V127 and MS–V128 from TS Table 3.3–9, and (4) make administrative changes to the TS that correct errors, remove an outdated license condition and an inconsistency between indexes and revises a reporting requirement. 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 ability of any operable structure, system, or component to perform its designated safety function is unaffected by this change. The proposed change neither installs or removes any plant equipment, nor alters the design, physical configuration, or mode of operation of any plant structure, system, or component. No physical changes are being made to the plant, so no new accident causal mechanisms are being introduced. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The margin of safety associated with the acceptance criteria of any accident is unchanged. The proposed change will have no affect on the availability, operability, or performance of safety-related systems and components. The proposed change relocates an existing limitation from the UFSAR to the technical specifications regarding availability of the charging pumps during operation in Mode 4 with the temperature of any RCS cold leg is less than or equal to 290 °F, in Mode 5, and in Mode 6 with the reactor vessel head on and the vessel head closure bolts not fully de-tensioned. Nonetheless, imposing this limitation does not alter the configuration or operation of the charging pumps from those specified in current administrative controls and the UFSAR. The proposed change includes revising the minimum water level in the service water system pump house required for operability of the service water system. This change replaces a non-conservative, incorrect value in the TS with a minimum required water level that is consistent with the design basis for the system. The elimination of MS–V127 and MS–V128 from TS Table 3.3–9 does not alter the design, configuration, operation, or function of these valves with regard to operation of the EFW system because in the existing design these normally open valves are not required to re-position to support operation of the TDEFW pump. Automatic valves MS–V393 and MS–V394, which actuate to initiate operation of the TDEFW pump, are appropriately under the control of TS Table 3.3–9. Last, the proposed amendment makes administrative changes to the TS that correct errors, remove an outdated license condition and an inconsistency between indexes and revises a reporting requirement.

The proposed changes do 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. Therefore, the margin of safety as defined in the TS is not reduced and the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Harold K. Chernoff.

Pacific Gas and Electric Co., Docket No. 50–133, Humboldt Bay Power Plant (HBPP), Unit 3 Humboldt County, California.

Date of amendment request: April 4, 2007.

Description of amendment request: The licensee has proposed amending the existing license to allow the results of near-term surveys, performed on a portion of the plant site, to be included in the eventual Final Status Survey (FSS) for license termination.

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

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

Response: No.

The proposed change would allow survey results for a specific area within the licensed site area, performed prior to Humboldt Bay Power Plant (HBPP) Unit 3 decommissioning and dismantlement activities, to be used in the overall licensed site area Final Status Survey (FSS) for license termination. The FSS will be performed following completion of HBPP Unit 3 decommissioning and dismantlement activities. This proposed change would not change plant systems or accident analysis, and as such, would not affect initiators of analyzed events or assumed mitigation of accidents. Therefore, the proposed change does not increase the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve a physical alteration to the plant or require existing equipment to be operated in a manner different from the present design. Implementation of a cross contamination prevention and monitoring plan will be done in accordance with plant procedures and licensing bases documents.

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

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

Response: No.

The proposed change has no effect on existing plant equipment, operating practices, or safety analysis assumptions. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mr. Antonio Fernandez, Esquire, Pacific Gas & Electric Company, Post Office Box 7442, San Francisco, CA 94120.

NRC Acting Branch Chief: Kristina Banovac.

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

Date of amendment request: April 17, 2007.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TSs) and license to establish more effective and appropriate action, surveillance, and administrative requirements related to ensuring the habitability of the control room envelop (CRE) in accordance with Nuclear Regulatory Commission (NRC) approved TS Task Force (TSTF) Standard Technical Specification change traveler TSTF–448, Revision 3, ''Control Room Habitability.'' The NRC staff issued a ''Notice of Availability of Technical Specification Improvement to Modify Requirements Regarding Control Room Envelope Habitability Using the Consolidated Line Item Improvement Process'' associated with TSTF–448, Revision 3, in the **Federal Register** on January 17, 2007 (72 FR 2022). The notice included a model safety evaluation, a model no significant hazards consideration (NSHC) determination, and a model license amendment request. In its application dated April 17, 2007, the licensee affirmed the applicability of the model NSHC determination which is presented below.

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

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

The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change revises the TS for the CRE emergency ventilation system, which is a mitigation system designed to minimize unfiltered air leakage into the CRE and to filter the CRE atmosphere to protect the CRE occupants in the event of accidents previously analyzed. An important part of the CRE emergency ventilation system is the CRE boundary. The CRE emergency ventilation system is not an initiator or precursor to any accident previously evaluated. Therefore, the probability of any accident previously evaluated is not increased. Performing tests to verify the operability of the CRE boundary and implementing a program to assess and maintain CRE habitability ensure that the CRE emergency ventilation system is capable of adequately mitigating radiological consequences to CRE occupants during accident conditions, and that the CRE emergency ventilation system will perform as assumed in the consequence analyses of design basis accidents. Thus, the consequences of any accident previously evaluated are not increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not impact the accident analysis. The proposed change does not alter the required mitigation capability of the CRE emergency ventilation system, or its functioning during accident conditions as assumed in the licensing basis analyses of design basis accident radiological consequences to CRE occupants. No new or different accidents result from performing the new surveillance or following the new program. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a significant change in the methods governing normal plant operation. The proposed change does not alter any safety analysis assumptions and is consistent with current plant operating practice. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The proposed change does not affect safety analysis acceptance criteria. The proposed change will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without compensatory measures. The proposed change does not adversely affect systems that respond to safely shut down the plant and to maintain the plant in a safe shutdown condition. 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: Harold K. Chernoff.

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: April 15, 2007.

Description of amendment request: The proposed amendments would modify the Technical Specifications (TSs) and license to establish more effective and appropriate action, surveillance, and administrative requirements related to ensuring the habitability of the control room envelop (CRE) in accordance with Nuclear Regulatory Commission (NRC) approved TS Task Force (TSTF) Standard Technical Specification change traveler TSTF–448, Revision 3, ''Control Room Habitability.'' The NRC staff issued a ''Notice of Availability of Technical Specification Improvement to Modify Requirements Regarding Control Room Envelope Habitability Using the Consolidated Line Item Improvement Process'' associated with TSTF–448, Revision 3, in the **Federal Register** on January 17, 2007 (72 FR 2022). The notice included a model safety evaluation, a model no significant hazards consideration (NSHC) determination, and a model license amendment request. In its application dated April 15, 2007, the licensee affirmed the applicability of the model NSHC determination which is presented below.

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

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

The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change revises the TS for the CRE emergency ventilation system, which is a mitigation system designed to minimize unfiltered air leakage into the CRE and to filter the CRE atmosphere to protect the CRE occupants in the event of accidents previously analyzed. An important part of the CRE emergency ventilation system is the CRE boundary. The CRE emergency ventilation system is not an initiator or precursor to any accident previously evaluated. Therefore, the probability of any accident previously evaluated is not increased. Performing tests to verify the operability of the CRE boundary and implementing a program to assess and maintain CRE habitability ensure that the CRE emergency ventilation system is capable of adequately mitigating radiological consequences to CRE occupants during accident conditions, and that the CRE emergency ventilation system will perform as assumed in the consequence analyses of design basis accidents. Thus, the consequences of any accident previously evaluated are not increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not impact the accident analysis. The proposed change does not alter the required mitigation capability of the CRE emergency ventilation system, or its functioning during accident conditions as assumed in the licensing basis analyses of design basis accident radiological consequences to CRE occupants. No new or different accidents result from performing the new surveillance or following the new program. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a significant change in the methods governing normal plant operation. The proposed change does not alter any safety analysis assumptions and is consistent with current plant operating practice. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The proposed change does not affect safety analysis acceptance criteria. The proposed change will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without compensatory measures. The proposed change does not adversely affect systems that respond to safely shut down the plant and to maintain the plant in a safe shutdown condition. 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: Harold K. Chernoff.

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

Date of amendment requests: March 30, 2007.

Description of amendment requests: The proposed amendment revises Technical Specifications (TSs) 3.8.1, ''AC [alternating current] Sources—Operating,'' 3.8.4, ''DC [direct current] Sources—Operating,'' 3.8.5, ''DC Sources—Shutdown,'' 3.8.6, ''Battery Cell Parameters,'' 3.8.7, ''Inverters—Operating,'' and 3.8.9, ''Distribution Systems—Operating.'' This change will also add a new Battery Monitoring and Maintenance Program, Section 5.5.2.16. The proposed TS changes will provide operational flexibility supported by DC electrical subsystem design upgrades that are in progress. These upgrades will provide increased capacity batteries, additional battery chargers, and the means to cross-connect DC subsystems while meeting all design battery loading requirements. With these modifications in place, it will be feasible to perform routine surveillances as well as battery replacements online.

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

Response: No.

The proposed changes to Technical Specifications (TS) 3.8.4 and 3.8.6 would allow extension of the Completion Time (CT) for inoperable Direct Current (DC) distribution subsystems to manually cross-connect DC distribution buses of the same safety train of the operating unit for a period of 30 days. Currently the CT only allows for 2 hours to ascertain the source of the problem before a controlled shutdown is initiated. Loss of a DC subsystem is not an initiator of an event. However, complete loss of a Train A (subsystems A and C) or Train B (subsystems B and D) DC system would initiate a plant transient/plant trip.

Operation of a DC Train in cross-connected configuration does not affect the quality of DC control and motive power to any system. Therefore, allowing the cross-connect of DC distribution systems does not significantly increase the probability of an accident previously evaluated in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR).

The above conclusion is supported by Probabilistic Risk Assessment (PRA) evaluation which encompasses all accidents, including UFSAR Chapter 15. The Frequency for Surveillance Requirements in TS 3.8.4.3 is changed from 24 months to 30 months. San Onofre Nuclear Generating Station (SONGS) experience has indicated that there have been no battery failures using the 24-month test frequency for battery service tests, and extending the interval to 30 months is not expected to affect SONGS' capability to detect battery health and capacity. Also, the routine test frequency of 30 months will better dove-tail with the scheduling of the more rigorous 60-month interval battery performance of modified performance discharge tests.

Enhancements from TSTF–360, Rev. 1 and IEEE 450 have been incorporated into Limiting Conditions for Operation (LCOs) 3.8.4, 3.8.5, and 3.8.6. These changes do not impact the probability or consequences of an accident previously evaluated.

Further changes are made of an editorial nature or provide clarification only. For example, discussions regarding electrical 'Trains' and 'Subsystems' will be in more conventional terminology. LCOs affected by editorial changes include 3.8.1, 3.8.4, 3.8.5, 3.8.6, 3.8.7, and 3.8.9.

The changes being proposed in the TS do not affect assumptions contained in other safety analyses or the physical design of the plant, nor do they affect other Technical Specifications that preserve safety analysis assumptions.

Therefore, 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 analyzed.

2. Will operation of the facility in accordance with this proposed change create the possibility of [a] new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change modifies surveillances and LCOs for batteries and chargers to meet the requirements of IEEE 450–2002 whose intent is to maintain the same equipment capability as previously assumed in our commitment to IEEE 450–1980.

The proposed change will allow the cross-tie of DC subsystems and allow extension of the CT for an inoperable subsystem to 30 days. Failure of the cross-tied DC buses and/or associated battery(ies) is bounded by existing evaluations for the failure of an entire electrical train.

Swing battery chargers are added to increase the overall DC system reliability. Administrative and mechanical controls will be in place to ensure the design and operation of the DC systems continue to meet the UFSAR design basis.

LCOs 3.8.1, 3.8.4, 3.8.5, 3.8.6, 3.8.7, and 3.8.9 revisions are editorial clarifications and do not affect plant design.

Therefore, operation of the facility in accordance with this proposed change will not create the possibility of [a] new or different kind of accident from any accident previously evaluated.

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

Response: No.

Changes in accordance with IEEE 450 and TSTF–360, Rev. 1 maintain the same level of equipment performance stated in the UFSAR and the current Technical Specifications.

Swing battery chargers are added to increase the overall DC system reliability. Administrative and mechanical controls will be in place to ensure the design and operation of the DC systems continue to meet the UFSAR design basis.

The addition of the DC cross-tie capability proposed for LCO 3.8.4 has been evaluated, as described previously, using PRA and determined to be of acceptable risk as long as the duration while cross-tied is limited to 30 days. An LCO has been included as part of this proposed change to ensure that plant operation, with DC buses cross-tied, will not exceed 30 days.

All remaining changes are editorial.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Thomas G. Hiltz.

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

Date of amendment request: April 25, 2007.

Description of amendment request: The proposed amendment would revise the technical specifications to increase the maximum number of tritium producing burnable absorber rods (TPBARs) that can be irradiated in the reactor from 240 to 400.

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

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

Response: No.

The proposed change modifies the maximum number of TPBARs in the core. The required boron concentration for the cold leg accumulators (CLAs) and RWST [Refueling Water Storage Tank] remains unchanged. The current boron concentration has been demonstrated to maintain the required accident mitigation safety function for the CLAs and RWST with the higher number of TPBARs and this will be verified for each core that contains TPBARs as part of the normal reload analysis. The CLAs and RWST safety function is to mitigate accidents that require the injection of borated water to cool the core and to control reactivity. These functions are not potential sources for accident generation and the modification of the number of TPBARs will not increase the potential for an accident. Therefore, the possibility of an accident is not increased by the proposed changes. The current boron concentration levels are supported by the proposed number of TPBARs in the core. Since the current boron concentration levels will continue to maintain the safety function of the CLAs and RWST in the same manner as currently approved, the consequences of an accident are not increased by the proposed changes.

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

Response: No.

The proposed change only modifies the maximum number of TPBARs in the core. The boron concentrations for accident mitigation functions of the CLAs and RWST remain unchanged. These functions do not have a potential to generate accidents as they only serve to perform mitigation functions associated with an accident. The proposed modification will maintain the mitigation function in an identical manner as currently approved. There are no plant equipment or operational changes associated with the proposed revision. Therefore, since the CLA and RWST functions are not altered and the plant will continue to operate without change, the possibility of a new or different kind of an accident is not created.

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

Response: No.

This change proposes a change to the maximum number of TPBARs in the core. The boron concentration requirements that support the accident mitigation functions of the CLAs and RWST remain unchanged. The proposed change does not alter any plant equipment or components and does not alter any setpoints utilized for the actuation of accident mitigation system or control functions. The proposed number of TPBARs, in conjunction with the current boron concentration values, has been demonstrated to provide an adequate level of reactivity control for accident mitigation and this will be verified for each core that contains TPBARs as part of the normal reload analysis. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: Thomas H. Boyce.

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 51.22(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.*

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Units Nos. 1, 2, and 3, Maricopa County, Arizona

Date of application for amendments: August 16, 2006, as supplemented by letters dated January 25 and March 8, 2007.

Brief description of amendments: The amendments revised Technical Specifications (TS) requirements in Surveillance Requirements (SRs) to allow for surveillances to be performed in modes that are not currently allowed in TS and to require certain SRs to be performed at a power factor of ≤0.89 if performed with the emergency diesel generators synchronized to the grid unless grid conditions do not permit.

Date of issuance: May 16, 2007.

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

Amendment Nos.: Unit 1—167, Unit 2—167, Unit 3—167.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Operating License and Technical Specifications.

Date of initial notice in ***Federal Register:*** October 24, 2006 (71 FR 62307). The supplements dated January 25 and March 8, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register** on October 24, 2006 (71 FR 62307).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 16, 2007.

No significant hazards consideration comments received: No.

Dominion Energy Kewaunee, Inc. Docket No. 50–305, Kewaunee Power Station, Kewaunee County, Wisconsin

Date of application for amendment: June 28, 2006, as supplemented by letter dated November 2, 2006.

Brief description of amendment: The amendment changes Kewaunee Power Station Technical Specifications 3.3.b.3.B and 3.3.b.4.A to increase the minimum required boron concentration in the refueling water storage tank from 2400 parts per million (ppm) to 2500 ppm.

Date of issuance: May 18, 2007.

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

Amendment No.: 192.

Facility Operating License No. DPR–43: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 1, 2006 (71 FR 43530). The supplemental letter contained clarifying information and did not change the initial no significant hazards consideration determination, and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 18, 2007.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of application for amendment: January 26, 2006, as supplemented by letter dated December 21, 2006.

Brief description of amendment: The amendment will allow additional startup and operating flexibility and an expanded operating domain resulting from the proposed implementation of the Average Power Range Monitor, Rod Block Monitor Technical Specification improvement program concurrently with the proposed implementation of the Maximum Extended Operating Domain Analysis, which is the combination of the power/flow operating map expansion with Maximum Extended Load Line Limit Analysis and increased core flow.

Date of issuance: May 17, 2007.

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

Amendment No.: 287.

Facility Operating License No. DPR–59: The amendment revised the License and the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 14, 2006 (71 FR 13171). The supplemental letter dated December 21, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 17, 2007.

No significant hazards consideration comments received: No.

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

Date of amendment request: November 2, 2006.

Description of amendment request: The proposed amendment revised Technical Specifications requirements for inoperable snubbers consistent with the Technical Specification Task Force 372, Revision 4.

Date of issuance: May 14, 2007.

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

Amendment No.: 229.

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

*Date of initial notice in **Federal Register:*** January 30, 2007 (72 FR 4307). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 14, 2007.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: August 31, 2006, as supplemented by letter dated January 31, 2007.

Brief description of amendment: The amendment relocated TS 3.8.7 requirements associated with 120 volt (V) inverter Y–28 and TS 3.8.9 requirements associated with the 120 V alternating current electrical power distribution subsystem panel C–540 to the Technical Requirements Manual.

Date of issuance: May 15, 2007.

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

Amendment No.: 230.

Renewed Facility Operating License No. DPR–51: Amendment revised the Technical Specifications/license.

*Date of initial notice in **Federal Register:*** November 7, 2006 (71 FR 65142). The supplement dated January 31, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 2007.

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: August 2, 2006.

Brief description of amendment: The amendment change deletes the augmented testing requirement for containment purge supply and exhaust isolation valves with resilient seal materials and allows the surveillance intervals to be set in accordance with the Containment Leakage Rate Testing Program.

Date of issuance: May 23, 2007.

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

Amendment No.: 213.

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

*Date of initial notice in **Federal Register:*** September 26, 2006 (71 FR 56191). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 23, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 6, 2006.

Brief description of amendment: This amendment revised the Ventilation Filter Test Program (VFTP) in Technical Specification 5.5.7, to correct the flow rate units specified in the VFTP, from standard cubic feet per minute to cubic feet per minute.

Date of issuance: May 9, 2007.

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

Amendment No.: 143.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications and License.

*Date of initial notice in **Federal Register:*** August 29, 2006 (71 FR 51228).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 9, 2007.

No significant hazards consideration comments received: No.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of application for amendment: May 25, 2006, as supplemented by letters dated December 21, 2006, March 14, 2007, and March 30, 2007.

Brief description of amendment: The amendment revises the Technical Specification Steam Generator tube Surveillance Program to one modeled after Technical Specification Task Force (TSTF) Traveler TSTF–449, ''Steam Generator Tube Integrity.''

Date of issuance: May 16, 2007.

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

Amendment No.: 223.

Facility Operating License No. DPR–72: Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 29, 2006 (71 FR 51229). The supplements dated December 21, 2006, March 14 and 30, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 29, 2006, as supplemented by letter dated December 7, 2006, and February 12, 2007.

Brief description of amendment: The amendment revises Technical Specification 3.7.8, ''Service Water (SW) System,'' from an electrical train-based specification to a pump-based specification. Revisions to the Limiting Conditions for Operation, Required Actions, Completion Times, and Surveillance Requirements have been made to require a specific number of SW water pumps to be operable rather than SW trains.

Date of issuance: May 16, 2007.

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

Amendment No.: 102.

Renewed Facility Operating License No. DPR–18: Amendment revised the License and Technical Specifications.

*Date of initial notice in **Federal Register:*** November 7, 2006 (71 FR 65144).

The letters dated December 7, 2006, and February 12, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 16, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 24, 2006, as supplemented on February 15, 2007.

Brief description of amendment: The amendment revises the Virgil C. Summer Nuclear Station Technical Specifications and provides associated Bases that are modeled after Technical Specification Task Force (TSTF) traveler, TSTF–449, Revision 4, ''Steam Generator Tube Integrity.'' A notice of availability for this TS improvement using the consolidated line item improvement process was published in the **Federal Register** on May 6, 2005 (70 FR 24126).

Date of issuance: May 15, 2007.

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

Amendment No.: 179.

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

Date of initial notice in ***Federal Register:*** June 20, 2006 (71 FR 35458). The supplemental letter provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration. The Commission's related evaluation of the amendment is contained in a safety evaluation dated May 15, 2007.

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: February 2, 2007.

Brief description of amendments: The amendments revised the Technical Specifications Limiting Condition for Operation (LCO) 3.10.1 to be consistent with TSTF–484, Revision 0, ''Use of Technical Specification 3.10.1 for Scram Time Testing Activities.''

Date of issuance: May 17, 2007.

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

Amendment Nos.: 251, 195.

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

Date of initial notice in ***Federal Register:*** March 13, 2007 (72 FR 11395).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 17, 2007.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 25th day of May 2007.

For the Nuclear Regulatory Commission.

Timothy J. McGinty,

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

[FR Doc. E7–10590 Filed 6–4–07; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Notice of Availability of Model Safety Evaluation and Model License Amendment Request on Technical Specification Improvement Regarding Relocation of Departure From Nucleate Boiling Parameters to the Core Operating Limits Report for Combustion Engineering Pressurized Water Reactors Using the Consolidated Line Item Improvement Process

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability.

SUMMARY: Notice is hereby given that the staff of the U.S. Nuclear Regulatory Commission (NRC) has prepared a model license amendment request (LAR), model safety evaluation (SE), and model proposed no significant hazards consideration (NSHC) determination related to changes to Standard Technical Specifications (STSs) for Combustion Engineering Pressurized Water Reactors (PWRs), NUREG–1432, Revision 3.1. This change allows the numerical limits located in technical specification (TS) 3.4.1, ''RCS Pressure, Temperature, and Flow [Departure from Nucleate Boiling (DNB)] Limits'' to be replaced with references to the Core Operating Limits Report (COLR). Associated changes are also included for the TS 3.4.1 Bases, and TS 5.6.3 ''Core Operating Limits Report (COLR).'' The Technical Specifications Task Force (TSTF) proposed these changes to the TS in TSTF–487 Revision 0, ''Relocate DNB Parameters to the COLR.'' This request was slightly modified in TSTF–487 Revision 1 on May 4, 2007.

The purpose of the model SE, LAR, and NSHC is to permit the NRC to efficiently process amendments to incorporate these changes into plant-specific TSs for Combustion Engineering PWRs. Licensees of nuclear power reactors to which the models apply can request amendments conforming to the models. In such a request, a licensee should confirm the applicability of the model LAR, model SE and NSHC determination to its plant.

DATES: The NRC staff issued a **Federal Register** Notice (72 FR 12223, March 15, 2007) which provided a model LAR, model SE, and model NSHC for comment related to replacing the DNB parameters in TS 3.4.1 with references to the COLR. The revised model LAR, revised model SE, and unchanged NSHC associated with this change are provided in this notice. The NRC can most efficiently consider applications based upon the model LAR, which references the model SE, if the application is submitted within one year of this **Federal Register** Notice.

FOR FURTHER INFORMATION CONTACT: William Cartwright, Mail Stop: O–12H2, Division of Inspection and Regional Support, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone (301) 415–8345.

SUPPLEMENTARY INFORMATION:

Background

This change was made using the Consolidated Line Item Improvement Process [CLIIP] for STS Changes for Power Reactors, issued on March 20, 2000 as Regulatory Information Summary 2000–006. This document can be viewed on the NRC's public Web page at *http://www.nrc.gov/reading-rm/doc-collections/gen-comm/reg-issues/2000/ri00006.html.* The CLIIP is intended to improve the efficiency and transparency of NRC licensing processes by processing proposed changes to the STS in a manner that supports subsequent license amendment applications. Those licensees opting to apply for the subject change to TSs are responsible for reviewing the NRC staff's evaluation, referencing the applicable technical justifications, and providing any necessary plant-specific information. This notice finalizes the model LAR and model SE. Each amendment application made in response to the notice of availability will be processed and noticed in accordance with applicable NRC rules and procedures.

The purpose of this change is to allow Combustion Engineering PWR licensees to recalculate cycle specific departure from nucleate boiling (DNB) parameter limits in the COLR using NRC-approved