receiving this Commission meeting schedule electronically, please send an electronic message to *dkw@nrc.gov.*

Dated: April 22, 2004.

Dave Gamberoni,

Office of the Secretary.

[FR Doc. 04–9596 Filed 4–23–04; 11:20 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

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 April 2, 2004, through April 15, 2004. The last biweekly notice was published on April 13, 2004 (69 FR 19561).

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 Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC 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, Ocean County, New Jersey

Date of amendment request: March 19, 2004.

Description of amendment request: The licensee proposed to revise Section 4.2, ''Reactivity Control,'' of the Technical Specifications. Specifically, the amendment would revise Subsection 4.2.C, regarding surveillance requirements associated with control rod scram time testing (STT) by: (1) Eliminating unnecessary depressurized STT of non-maintenance-affected control rods, (2) providing the required STT data necessary to apply actual scram times to implement improved minimum critical power ratio operating limits, and (3) eliminating the resulting redundant requirement to test ''eight control rods'' after a reactor scram or other outage. The amendment will also include editorial and pagination changes to accommodate the proposed technical changes.

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

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

The proposed change adds new surveillance requirements (SR) to the Minimum Critical Power Ratio (MCPR) Technical Specification (TS) which requires determination of the MCPR operating limit following the completion of scram time testing (STT) of the control rods. Use of the scram speed in determining the MCPR operating limit (*i.e.,* Option B) is an alternative to the current method for determining the operating limit (*i.e.,* Option A). The probability of an accident previously evaluated is unrelated to the MCPR operating limit that is provided to ensure no fuel damage results during anticipated operational occurrences. This is an operational limit to ensure conditions following an assumed accident do not result in fuel failure and therefore do not contribute to the occurrence of an accident. The proposed change eliminates unnecessary depressurized STT of non-maintenance[-]affected control rods and the requirement to test ''eight selected rods'' after a reactor scram or other outage. The requirement to test ''eight selected rods'' is replaced by a new SR to perform periodic STT. No active or passive failure mechanisms that could lead to an accident are affected by this proposed change. Therefore, the proposed change in STT requirements does not significantly increase the probability of an accident previously evaluated.

The proposed change ensures that the appropriate MCPR operating limit is in place. By implementing the correct MCPR operating limit the MCPR safety limit will continue to be ensured. Ensuring the MCPR safety limit is not exceeded will result in prevention of fuel failure. Therefore, since there is no increase in the potential for fuel failure there is no increase in the consequences of any accidents previously evaluated.

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

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

Response: No.

The proposed change adds a new SR to the MCPR TS which requires determination of the MCPR operating limit following the completion of scram time testing of the control rods. The proposed change eliminates unnecessary depressurized STT of non-maintenance[-]affected rods and the requirement to test ''eight selected rods'' after a reactor scram or other outage. The requirement to test ''eight selected rods'' is replaced by a new SR to perform periodic STT. The proposed change does not involve the use or installation of new equipment. Installed equipment is not operated in a new or different manner. No new or different system interactions are created, and no new processes are introduced. No new failures have been created by the addition of the proposed SR and the use of the alternate method for determining the MCPR operating limit.

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.

Use of Option B for determining the MCPR operating limit will result in a reduced operating limit in comparison to the use of Option A. However, a reduction in the operating limit margin does not result in a reduction in the safety margin. The MCPR safety limit remains the same regardless of the method used for determining the operating limit. The proposed change eliminates unnecessary depressurized STT of non-maintenance[-]affected control rods and the requirement to test ''eight selected rods'' after a reactor scram or other outage. The requirement to test ''eight selected rods'' is replaced by a new SR to perform periodic STT. No active or passive failure mechanisms that could adversely impact the consequences of an accident are affected by this proposed change. All analyzed transient results remain well within the design values for structures, systems and components.

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: March 23, 2004.

Brief description of amendments: The licensee proposed to revise the Technical Specifications (TSs) by eliminating the requirements for hydrogen/oxygen monitors. The proposed amendment supports implementation of the revision to 10 CFR 50.44, ''Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors,'' that became effective on October 16, 2003.

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–447, ''Elimination of Hydrogen Recombiners and Change to Hydrogen and Oxygen Monitors.'' The availability of this TS improvement was published in the **Federal Register** on September 25, 2003 (68 FR 55416), on possible amendments concerning TSTF–447, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. In its application for amendment, the licensee affirmed the applicability of the following NSHC determination.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee presented an analysis of NSHC by endorsing the model NSHC determination published in 68 FR 55416 (reproduced below):

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

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

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

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

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

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

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

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

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

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

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

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

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

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

The NRC staff has reviewed the 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, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Section Chief: Richard J. Laufer.

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

Date of amendments request: November 12, 2002, as supplemented March 5, 2004. This notice supersedes the notice that was published on February 18, 2003 (68 FR 7813).

Description of amendments request: The proposed amendments would revise the Technical Specifications to support an expansion of the core flow operating range, including the new automated backup stability protection function.

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 will implement DSS–CD [Detect and Suppress Solution—Confirmation Density] as the long-term stability solution. The DSS–CD solution is designed to identify the power oscillation upon inception and initiate control rod insertion to terminate the oscillations prior to any significant amplitude growth. The DSS–CD provides protection against violation of the Safety Limit Minimum Critical Power Ratio (SLMCPR) for anticipated oscillations. Compliance with General Design Criteria (GDC) 10 and 12 of 10 CFR part 50, Appendix A is accomplished via an automatic action. The DSS–CD introduces an enhanced detection algorithm that detects the inception of power oscillations and generates an earlier power suppression trip signal exclusively based on successive period confirmation recognition. The existing Option III algorithms are retained, with generic setpoints, to provide defense-in-depth protection for unanticipated reactor instability events.

A developing instability event is suppressed by the DSS-CD system with substantial margin to the SLMCPR and no clad damage, with the event terminating in a scram and never developing into an accident. In addition, the DSS–CD solution defense-in-depth features incorporate all the backup scram algorithms plus the licensed scram feature of the existing Option III system. The DSS–CD system does not interact with equipment whose failure could cause an accident. Scram setpoints in the DSS–CD will be established so that analytical limits are met. The reliability of the DSS–CD will meet or exceed that of the existing system. No new challenges to safety-related equipment will result from the DSS–CD solution. Because an instability event would reliably terminate in an early scram without impact on other safety systems, there is no significant increase in the probability of an accident.

The existing requirement to initiate an alternate (*i.e.*, manual) method to detect and suppress thermal hydraulic instability oscillations is expanded to include a requirement to either implement an Automated Backup Stability Protection (ABSP) (*i.e.*, Required Action I.2.1) or exit the operating region most susceptible to rapid onset of Thermal Hydraulic Instability (THI) (*i.e.*, Required Action I.2.2). The ABSP is an automatic reactor scram region, implemented by the Average Power Range Monitor (APRM) flow-biased scram setpoint. It may be used if the Oscillation Power Range Monitoring (OPRM) system is inoperable to allow continued operation within the MELLLA+ [Maximum Extended Load Line Limit Analysis Plus] operating domain. Additionally, a new Required Action I.3 is included. Required Action I.3 ensures that a report is made to the NRC, if DSS–CD is inoperable for 120 days.

To maintain the existing margin between equipment operability requirements and the region of power-flow operation where anticipated events could lead to thermal-hydraulic instability, (1) TS 3.3.1.1, Required Action J.1 is revised to require the plant to be < 18% RTP [rated thermal power] versus < 20% RTP in the event that the OPRM Upscale Function is inoperable and the Required Actions associated with Action I are not completed, and (2) the operability requirement for the OPRM Upscale Function (*i.e.*, TS 3.3.1.1, Table Function 2.f) is changed from ≥ 20% RTP to ≥ 18% RTP. This 5% margin is consistent with and maintains the existing 5% margin operability requirements for the Option III OPRM Upscale operability requirements.

Overall, these changes result in more conservative plant operation. Other changes proposed in this supplement are either in direct support of ABSP or are administrative in nature.

Proper operation of the DSS–CD system does not affect any fission product barrier or Engineered Safety Feature. Thus, the proposed change cannot change the consequences of any accident previously evaluated. As stated above, the DSS–CD solution meets the requirements of GDC 10 and 12 by automatically detecting and suppressing design basis thermal-hydraulic oscillations prior to exceeding the fuel SLMCPR.

Based on the above, the operation of the DSS–CD solution within the framework of the Option III OPRM hardware will not increase 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 DSS–CD solution operates within the existing Option III OPRM hardware. No new operating mode, safety-related equipment lineup, accident scenario, system interaction, or equipment failure mode was identified. The ABSP automatic reactor scram region is implemented by adjusting the existing APRM flow-biased scram setpoint. Therefore, the DSS–CD solution will not adversely affect plant equipment.

Because there are no hardware changes, there is no change in the possibility or consequences of a failure. The worst case failure of the equipment is a failure to initiate mitigating action (*i.e.*, scram), but no failure can cause an accident of a new or different kind than any previously evaluated.

Based on the above, the proposed change to the DSS–CD solution 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.

The DSS–CD solution is designed to identify the power oscillation upon inception and initiate control rod insertion to terminate the oscillations prior to any significant amplitude growth. The DSS–CD solution algorithm will maintain or increase the margin to the SLMCPR for anticipated instability events. The safety analyses in ''Detect And Suppress Solution—Confirmation Density Licensing Topical Report,'' Revision 3 demonstrate the margin to the SLMCPR for postulated bounding stability events. Existing margin between equipment operability requirements and the region of power-flow operation where anticipated events could lead to thermal-hydraulic instability are maintained. As a result, there is no impact on the SLMCPR identified for an instability event.

The existing requirement to initiate an alternate method to detect and suppress thermal hydraulic instability oscillations is expanded to include a requirement to either implement an ABSP (*i.e.*, Required Action I.2.1) or exit the operating region most susceptible to rapid onset of THI (*i.e.*, Required Action I.2.2). Additionally, a new Required Action I.3 is included. Required Action I.3 ensures that a report is made to the NRC, if DSS–CD is inoperable for 120 days. These change results in more conservative plant operation. Other changes proposed in this supplement are either in direct support of ABSP or are administrative in nature.

The current Option III algorithms (*i.e.*, Period Based Detection, Amplitude Based, and Growth Rate) are retained (with generic setpoints) to provide defense-in-depth protection for unanticipated reactor instability events.

Based on the above, the proposed change will not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Steven R. Carr, Associate General Counsel—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: William F. Burton, Acting.

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

Date of amendment request: March 25, 2004.

Description of amendment request: The proposed amendments would allow the use of the methodology described in Framatome-ANP (FRA–ANP) Topical BAW–10169–A ''RSG Plant Safety Analysis—B&W Safety Analysis Methodology for Recirculating Steam Generator Plants'', dated October 1989 for the generation of mass and energy release rates during a Main Steam Line Break accident for Prairie Island Nuclear Generating Plant.

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

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

Response: No.

The proposed amendment will change the Prairie Island Nuclear Generating Plant licensing basis by allowing the use of the methodology described in Framatome-ANP Topical BAW–10169–A ''RSG Plant Safety Analysis—B&W Safety Analysis Methodology for Recirculating Steam Generator Plants'' that utilizes the RELAP5/MOD2–B&W code described in Topical BAW–10164–A ''RELAP5/MOD2–B&W—An Advanced Computer Program for Light-Water Reactor LOCA [loss-of-coolant accident] and Non-LOCA Transient Analysis'' for the generation of predicted mass and energy releases during a Main Steam Line Break accident.

The methodology used to perform an analysis of a main steam line break is not an accident initiator, thus changing the methodology does not increase the probability of an accident.

The mass and energy releases generated by the proposed methodology will be utilized to demonstrate that the design basis limits for fission product barriers are not exceeded. The proposed methodology does not alter the nuclear reactor core, reactor coolant system, or equipment used directly in mitigation of a main steam line break, thus radioactive releases due to a main steam line break accident are not affected by the proposed change in analysis methodology. Therefore, this change does not increase the consequences of an accident previously evaluated.

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

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

Response: No.

The proposed amendment will change the Prairie Island Nuclear Generating Plant licensing basis by allowing the use of the methodology described in Framatome-ANP Topical BAW–10169–A ''RSG Plant Safety Analysis—B&W Safety Analysis Methodology for Recirculating Steam Generator Plants'' that utilizes the RELAP5/MOD2–B&W code described in Topical BAW–10164–A ''RELAP5/MOD2–B&W—An Advanced Computer Program for Light-Water Reactor LOCA and Non-LOCA Transient Analysis'' for the generation of predicted mass and energy releases during a Main Steam Line Break accident.

The analysis of a main steam line break using the proposed methodology does not alter the nuclear reactor core, reactor coolant system, or equipment used directly in mitigation of a main steam line break.

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

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

Response: No.

The proposed amendment will change the Prairie Island Nuclear Generating Plant licensing basis by allowing the use of the methodology described in Framatome-ANP Topical BAW–10169–A ''RSG Plant Safety Analysis—B&W Safety Analysis Methodology for Recirculating Steam Generator Plants'' that utilizes the RELAP5/MOD2–B&W code described in Topical BAW–10164–A ''RELAP5/MOD2–B&W—An Advanced Computer Program for Light-Water Reactor LOCA and Non-LOCA Transient Analysis'' for the generation of predicted mass and energy releases during a Main Steam Line Break accident.

The proposed licensing basis change will result in a conservative calculation of the mass and energy releases during a Main Steam Line Break accident. This will ensure that there is no reduction in the margin of safety for analyses that utilize the generated mass and energy releases as inputs. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: L. Raghavan.

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: March 4, 2004.

Description of amendment request: The proposed amendment would revise the SSES 1 and 2 Technical Specification Table 3.3.5.1–1 to clarify that four low pressure coolant injection pump discharge pressure-high channels are required for each automatic depressurization system trip function.

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 Technical Specification required number of protection channels is not an initiator to any accident sequence analyzed in the Final Safety Analysis Report (FSAR). As discussed in this request, the change is editorial and involves no change in the number of ADS [Automatic Depressurization System] supporting protection channels required by the Susquehanna Steam Electric Station (SSES) Technical Specifications (TS). The change does not have any effect on the initiator of any accident sequence analyzed in the Final Safety Analysis Report (FSAR) and does not affect any assumptions associated with the mitigation of accident or transient events. The change does not involve any physical change to structures, systems, or components (SSCs) and does not involve any physical change to structures, systems, or components (SSCs) and does not alter the method of operation or control of SSCs. The current assumptions in the SSES FSAR safety analysis regarding accident initiators and mitigation of accidents are unaffected by these changes. No additional failure modes or mechanisms are being introduced and the likelihood of previously analyzed failures remains unchanged.

Operation in accordance with the proposed Technical Specification (TS) continues to ensure that the plant response to analyzed accidents remains capable of performing as described in the FSAR. Therefore, the mitigative functions supported by the system continue to provide the protection assumed by the analysis.

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 equipment is being introduced, and installed equipment is not being operated in a new or different manner. There are no setpoints, at which protective or mitigative actions are initiated, affected by this change. This change does not alter the manner in which equipment operation is initiated, nor are the function demands on credited equipment be[ing] changed. No alterations in the procedures that ensure the plant remains within analyzed limits are being proposed, and no changes are being made to the procedures relied upon to respond to an off-normal event as described in the FSAR. As such, no new failure modes are being introduced. The change does not alter the assumptions made in the safety analysis and licensing basis.

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

Response: No.

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The change is editorial and involves no technical changes to the Susquehanna Steam Electric Station (SSES) Technical Specifications (TS). Therefore the plant response to analyzed events continues to provide the margin of safety assumed by the analysis.

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 Section Chief: Richard J. Laufer.

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

Date of amendment request: March 5, 2004.

Description of amendment request: The proposed amendment would revise Technical Specification Surveillance Requirement (SR) 3.6.4.1.3 to require that only one secondary containment access door in each access opening be verified closed. In addition, this SR allows entry and exit access between required secondary containment zones that have a single door.

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 Technical Specification Surveillance being revised, which verifies the status of the secondary containment access doors, is not an initiator to any accident sequence analyzed in the Final Safety Analysis Report (FSAR). The proposed change relaxes the acceptance criteria of this Surveillance such that maintenance on one of two airlock access doors can be performed. However, requiring that at least one door is closed, in conjunction with the continued requirement to maintain the building at a negative pressure, continues to assure that the secondary containment barrier is maintained operable. This provides adequate assurance that the secondary containment is capable of performing the accident mitigation function assumed in the accident analyses. As a result, the consequences of any accident previously evaluated are not significantly affected.

The Note, which was added to the Technical Specifications, provides clarification and precludes a conflict with the explicit wording of SR 3.6.4.1.3. Since this Note is consistent with the intent as reflected in the Bases and with the prior SSES Technical Specifications, the change is considered editorial and reflects an administrative presentation preference and not a technical change.

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

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

Response: No.

The proposed change does not involve a physical alteration of the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There are no setpoints, at which protective or mitigative actions are initiated, affected by this change. This change does not alter the manner in which equipment operation is initiated, nor are the function demands on credited equipment changed. No alterations in the procedures that ensure the plant remains within analyzed limits are being proposed, and no changes are being made to the procedures relied upon to respond to an off-normal event as described in the FSAR. As such, no new failure modes are being introduced.

The Note, which was added to the Technical Specifications, provides clarification and precludes a conflict with the explicit wording of SR 3.6.4.1.3. Since this Note is consistent with the intent as reflected in the Bases and with the prior SSES Technical Specifications, the change is considered editorial and reflects an administrative presentation preference and not a technical change.

The change does not alter the assumptions made in the safety analysis and licensing basis.

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

Response: No.

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The change could allow additional time for one of two airlock doors to be open for maintenance. However, the margin of safety is maintained by the continued closure of the remaining airlock door (as is currently allowed for normal entry and exit) and the continued requirement to be able to maintain the building at a negative pressure.

The Note, which was added to the Technical Specifications, provides clarification and precludes a conflict with the explicit wording of SR 3.6.4.1.3. Since this Note is consistent with the intent as reflected in the Bases and with the prior SSES Technical Specifications, the change is considered editorial and reflects an administrative presentation preference and not a technical change.

Therefore, the plant response to analyzed events continues to provide the margin of safety assumed by the analysis.

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 Section Chief: Richard J. Laufer.

Tennessee Valley Authority, Docket Nos. 50–259, 50–260 and 50–296, Browns Ferry Nuclear Plant (BFN), Units 2 and 3, Limestone County, Alabama

Date of amendment request: July 31, 2002, as supplemented by letters dated December 9, 2002, February 12, 2003, March 26, 2003, July 11, 2003, and July 17, 2003.

Description of amendment request: The proposed amendments request full implementation of an alternative source term (AST) for the Units 1, 2, and 3 operating licenses. The amendments adopt the AST methodology by revising the current accident source term and replacing it with an accident source term as prescribed in 10 CFR 50.67. The submittals also propose to revise/delete the Technical Specification (TS) Sections associated with control emergency ventilation (CREV), standby gas treatment (SGT), standby liquid control (SLC), and secondary containment systems. Additionally, the submittals request modification of the licensing and design basis to reflect the application of the AST methodology and the function of the SLC system, and deletion of a license condition for Units 2 and 3, which all the actions have been completed.

The supplements to the original application include the withdrawal of the request to delete one of the TS Sections described above, associated with the absorption of elemental iodine by the SGT and CREV systems charcoal filters. Also the supplements add a new TS Section to require verification that the minimum fuel decay period has passed prior to moving fuel after the reactor is shut down. The licensee indicated that these modifications/deletions do not affect the originally published no significant hazards consideration. The original no hazards consideration is reproduced below.

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:

A. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The AST and those plant systems affected by implementing AST do not initiate DBAs [design-basis accidents]. The AST does not affect the design or operation of the facility; rather, once the occurrence of an accident has been postulated, the new source term is an input to evaluate the consequences. The implementation of the AST has been evaluated in the analyses for the limiting DBAs at BFN. The equipment affected by the proposed change is mitigative in nature and relied upon following an accident. The proposed changes to the TS do revise certain performance requirements. However, these changes will not involve a revision to the parameters or conditions that could contribute to the initiation of a design basis accident discussed in Chapter 14 of the BFN Updated Final Safety Analysis Report.

Plant specific radiological analyses have been performed and, based on the results of these analyses, it has been demonstrated that the dose consequences of the limiting events considered in the analyses are within the regulatory guidance provided by the NRC for use with the AST. This guidance is presented in 10 CFR 50.67, Regulatory Guide 1.183, and Standard Review Plan Section 15.0.1. Therefore, the proposed amendment does not result in a significant increase in the consequences or a significant increase in the probability of any previously evaluated accident.

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

Implementation of AST does not alter any design basis accident initiators. These changes do not affect the design function or mode of operations of systems, structures, or components in the facility prior to a postulated accident. Since systems, structures, and components are operated essentially no differently after the AST implementation, no new failure modes are created by this proposed change. Therefore, the proposed license amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. The proposed amendment does not involve a significant reduction in a margin of safety.

The changes proposed are associated with a revision to the licensing basis for BFN. The results of accident analyses revised in support of the proposed change are subject to the acceptance criteria in 10 CFR 50.67. The analyzed events have been carefully selected, and the analyses supporting this submittal have been performed using approved methodologies. The dose consequences of these limiting events are within the acceptance criteria provided by the regulatory guidance as presented in 10 CFR 50.67, Regulatory Guide 1.183, and SRP 15.0.1.

Therefore, because the proposed changes continue to result in dose consequences within the applicable regulatory limits, the changes are considered to 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 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 Section Chief: William F. Burton (Acting).

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

Date of amendment request: March 3, 2004 (TSC 03–10).

Description of amendment request: The proposed amendment would revise the Updated Final Safety Analysis Report (UFSAR) and the Technical Specification Bases description of the seismic qualification of round flexible ducting, triangular ducting, and associated air bars installed as part of the suspended ceiling air delivery system in the main control room.

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 design function of the MCR [main control room] ducting system is to support pressurization and cooling of the control room during normal and accident conditions. The design function of the MCR suspended ceiling is to remain in place during and subsequent to an accident, support the triangular and flexible ducts, and not damage safety-related equipment. The MCR ducting, including the classification and methodology changes, is a passive feature and does not act as an accident initiator, *i.e.*, failure of the ducting would not initiate a design basis accident. The MCR suspended ceiling has been qualified such that it will remain in place and perform its safety function during and after an accident. Consequently, the changes associated with the MCR ducting and suspended ceiling do not affect the frequency of occurrence for accidents previously evaluated in the UFSAR.

For the principal design basis accidents, loss of coolant accident (LOCA), internal flood, steam generator tube rupture (SGTR), main steam line break (MSLB), etc., the integrity of the MCR HVAC [heating, ventilation and air conditioning] system, including the suspended ceiling, will not be compromised. These accidents do not have a structural effect on the MCR. This means that for radiological or toxic chemical accidents, the ability to both pressurize and maintain MCR temperatures within the design limits is unaffected by the limited quality and seismic requirements for the flexible and triangular ducting.

An accident that involves a fire that affects the MCR or the habitability of the MCR was not a consideration for the qualification of the air distribution components. A fire of this nature will result in plant operation from the Auxiliary Control Room (ACR) which is supported by a separate HVAC system.

The physical effects of an earthquake (including the design basis SSE) is the only event in which the design basis for the MCR HVAC is potentially challenged. An evaluation by an industry seismic expert shows that the ducting and suspended ceiling will remain in place, will retain their structural integrity such that flow will not be impeded, and the ducting pressure boundary will not be lost. Thus, reducing the QA [quality assurance] and seismic qualification requirements for the MCR ducting and changing the method of seismic qualification will not result in loss of safety function for any design basis accident or event. Thus, the accident dose as previously evaluated in the UFSAR is not affected by the proposed license amendment.

Based on the above discussion, 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?

No. The MCR ducting addressed by the proposed amendment is not an accident initiator; *i.e.*, failure of the ducting will not initiate a design basis accident. In addition, the subject ducting and suspended ceiling have been evaluated and a determination has been made that they will continue to perform their safety functions during normal and accident conditions. Consequently, this activity does not create a possibility of a new or different type of accident than any previously evaluated.

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

No. The changes addressed in TVA' s proposed amendment are associated with changes in QA requirements and seismic qualification methodology for safety related air delivery components and for the suspended ceiling. The change does not affect specific HVAC equipment safety limits, design limits, set points, or other critical parameters. In addition, the new seismic analysis methodology and limited QA requirements ensure that these components will continue to perform their safety functions during normal and accident conditions. The previously implied margin of safety against structural or functional failure of the air delivery components or suspended ceiling during and after a design basis SSE [safe-shutdown earthquake] has not been reduced. 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Section Chief: William F. Burton, Acting.

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

Date of amendment request: April 7, 2004.

Description of amendment request: The proposed amendment would revise the maximum ultimate heat sink (UHS) temperature by revising the Technical Specification (TS) maximum essential raw cooling water (ERCW) temperature limit.

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 change to increase the UHS maximum temperature will not adversely alter the function, design, or operating practices for plant systems or components. The UHS is utilized to remove heat loads from plant systems during normal and accident conditions. This function is not expected or postulated to result in the generation of any accident and continues to adequately satisfy the associated safety functions with the proposed changes. Therefore, the probability of an accident presently evaluated in the safety analyses will not be increased. With the exception of re-gearing the shutdown board room chiller compressors, no other plant equipment must be altered as a result of this change. Re-gearing of the shutdown board room chillers will ensure their continued performance in accordance with design concurrent with the increased UHS temperature. The heat loads that the UHS is designed to accommodate have been evaluated for functionality with the higher temperature limits. The result of these evaluations is that there is existing margin associated with the systems that utilize the UHS for normal and accident conditions. These margins are sufficient to accommodate the postulated normal and accident heat loads with the proposed changes to the UHS. Since the safety functions of the UHS are maintained, the systems that ensure acceptable offsite dose consequences will continue to operate as designed. The change in the maximum calculated containment pressure associated with the design basis loss of coolant accident remains below the ASME [American Society of Mechanical Engineers] Code design internal pressure. The change to clarify the maximum allowable internal containment pressure is administrative consistent with present wording in the TS Bases. Therefore, the consequence of any accident will be the same as those previously analyzed.

Therefore, since the UHS safety function will continue to meet accident mitigation requirements and limit dose consequences to acceptable levels, TVA has concluded that 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?

No. The UHS function provides accident mitigation capabilities and serves as a heat sink for normal and upset plant conditions; the UHS is not an initiator of any accident. By allowing the proposed change in the UHS temperature requirements, only the parameters for UHS operation are changed while the safety functions of the UHS and systems that transfer the heat sink capability continue to be maintained. The proposed change does not impact the response of the systems and components assumed in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The proposed change has been evaluated for systems that are needed to support accident mitigation functions as well as normal operational evolutions. Operational margins were found to exist in the systems that utilize the UHS capabilities such that these proposed changes will not result in the loss of any safety function necessary for normal or accident conditions. The ERCW system has excess flow margins that will accommodate the increased flows necessary for the proposed temperature increase. While operating margins have been reduced by the proposed changes, safety margins have been maintained as assumed in the accident analyses for postulated events. The proposed change results in an increase in the maximum calculated containment peak pressure. However, the change in the maximum calculated containment peak pressure associated with the design basis LOCA [loss-of-coolant accident] is a small percentage of the margin between the current maximum calculated containment peak pressure and the ASME Code design internal pressure. The change to clarify the maximum allowable internal containment pressure is administrative. This aspect of the proposed change does not involve a significant reduction in a margin of safety. Additionally, the proposed changes do not require any further modification (the shutdown board room chiller will be re-geared) of component setpoints or operating provisions that are necessary to maintain margins of safety established by the WBN design. 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Section Chief: William F. Burton, Acting.

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

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

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

STP Nuclear Operating Company, Docket No. 50–499, South Texas Project, Unit 2, Matagordo County, Texas

Date of amendment request: March 4, 2004.

Brief description of amendment request: The proposed amendment would allow South Texas Project (STP) Unit 2 to change modes with standby diesel generator 22 inoperable. This is a one-time change that would expire 14 days after entering Mode 4 on restart from the STP Unit 2 Spring 2002 refueling outage.

Date of publication of individual notice in ***Federal Register:*** March 23, 2004.

Expiration date of individual notice: April 22, 2004 (public comments), and May 24, 2004 (hearing requests).

STP Nuclear Operating Company, Docket No. 50–499, South Texas Project, Unit 2, Matagordo County, Texas

Date of amendment request: March 18, 2004.

Brief description of amendment request: These amendments revise Technical Specification (TS) Surveillance Requirement 4.7.7.e.3 to add a footnote that allows an evaluation for points that do not meet the 1/8 inch Water Gauge criterion of the current TS. These amendments close out Notice of Enforcement Discretion No. 04–6–001, which the Commission granted on March 23, 2004.

Date of publication of individual notice in ***Federal Register:*** April 5, 2004.

Expiration date of individual notice: April 19, 2004 (public comments), and June 4, 2004 (hearing requests).

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: March 23, 2004.

Description of amendments request: To allow both trains of control room air-conditioning system to be inoperable for up to 7 days, provided control room temperatures are verified every 4 hours to be less than or equal to 90 degrees Fahrenheit.

Date of publication of individual notice in the ***Federal Register:*** April 14, 2004 (69 FR 19880).

Expiration date of individual notice: May 14, 2004.

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, 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 NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov*.

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 application for amendments: July 14, 2003, as supplemented December 5, 2003, and February 12, 2004.

Brief description of amendments: These amendments change the Surveillance Requirement 3.6.6.8 to verify each containment spray nozzle is unobstructed only following maintenance that could result in nozzle blockage.

Date of issuance: April 8, 2004.

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

Amendment Nos.: 264 and 241.

Renewed Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 19, 2003 (68 FR 49814). The supplements dated December 5, 2003, and February 12, 2004, 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 these amendments is contained in a Safety Evaluation dated April 8, 2004.

No significant hazards consideration comments received: No.

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 application for amendments: February 25, 2004.

Brief description of amendments: These amendments changes the implementation date for the new cooldown rates for pressure temperature limits established by Amendment Nos. 261 and 238 for Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, respectively, from 120 days after issuance, to July 1, 2004.

Date of issuance: April 5, 2004.

Effective date: As of the date of issuance, immediately changing the implementation date of Amendment Nos. 261 and 238 to July 1, 2004.

Amendment Nos.: 263 and 240.

Renewed Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 5, 2004 (69 FR 10487). The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated April 5, 2004.

No significant hazards consideration comments received: No.

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of application for amendment: March 20, 2003; as supplemented on March 31, April 17, June 11, July 21, and December 11, 2003; and January 20, February 10, and March 11, 2004.

Brief description of amendment: The amendment revised the Technical Specifications (TSs) to reflect an expanded operating domain resulting from the implementation of the Average Power Range Monitor, Rod Block Monitor TSs/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA).

Date of Issuance: April 14, 2004.

Effective date: As of the date of issuance, and shall be implemented at the start of operating cycle 24.

Amendment No.: 219.

Facility Operating License No. DPR–28: Amendment revised the TS.

*Date of initial notice in **Federal Register:*** April 15, 2003 (68 FR 18276). The licensee's March 31, April 17, June 11, July 21, and December 11, 2003; and January 20, February 10, and March 11, 2004, letters 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 this amendment is contained in a Safety Evaluation dated April 14, 2004.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi; Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana; and Entergy Operations, Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of application for amendment: November 6, 2002, as supplemented by letters dated November 18, 2003, and January 30, 2004.

Brief description of amendment: The amendment would revise the Facility Operating Licenses, Appendix B, Environmental Protection Plan (EPP) (Non-Radiological) for the respective plants.

Date of issuance: April 12, 2004.

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

Amendment Nos: 165, Docket No. 50–416, NPF–29; 138, Docket No. 50–458, NPF–47; 193, Docket No. 50–382, NPF–38.

Facility Operating License Nos. NPF–29, NPF–47, and NPF–38: The amendments revise the EPPs for the respective plants.

*Date of initial notice in **Federal Register:*** December 10, 2002 (67 FR 75872).

The licensee enclosed a revised no significant hazards consideration (NSHC) determination with the supplemental letter dated November 18, 2003. This revised NSHC determination contained minor wording changes as compared with the NSHC determination included in the original application dated November 6, 2002, changes made to reflect the new EPP changes, and did not expand the scope of the application as originally noticed, and did not change the conclusions of the NSHC determination as published in the **Federal Register** on December 10, 2002 (67 FR 75872). The January 30, 2004, supplemental letter provided further clarification to the November 18, 2003, supplemental letter that did not change the conclusion of the NSHC determination published on December 10, 2002.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 12, 2004.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: June 20, 2003, as supplemented by letter dated December 12, 2003.

Brief description of amendment: The amendment authorizes changes to the surveillance requirements for containment integrated leak rate testing in TS 4.4.a, "Integrated Leak Rate Tests (Type A)."

Date of issuance: April 6, 2004.

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

Amendment No.: 173.

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

*Date of initial notice in **Federal Register:*** July 22, 2003 (68 FR 43391) . 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 April 6, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 27, 2003, as supplemented by letters dated October 30, and December 19, 2003.

Brief description of amendments: The proposed amendment would approve a selective scope application of an alternative source term for fuel-handling accidents. Specifically, the amendments would revise Technical Specification 3.9.3, "Containment Penetrations," to (1) change the Applicability statement to "During movement of recently irradiated fuel assemblies within containment," and (2) modify the Required Action for Condition A to eliminate the requirement to suspend core alterations and add the requirement to suspend movement of recently irradiated fuel assemblies within containment if one or more containment penetrations are not in the required status.

Date of issuance: April 2, 2004.

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

Amendment Nos.: 213 and 218.

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

*Date of initial notice in **Federal Register:*** May 13, 2003 (68 FR 25656). The supplemental letters 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 amendments is contained in a Safety Evaluation dated April 2, 2004.

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: February 25, 2003, as supplemented September 9, 2003.

Brief description of amendment: The amendment added an allowed-outage time for Engineered Safety Features Actuation System Instrumentation channels to be out of service in a bypassed state.

Date of issuance: April 5, 2004.

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

Amendment No.: 167.

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

*Date of initial notice in **Federal Register:*** April 1, 2003 (68 FR 15762). The September 9, 2003, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the application.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 5, 2004.

No significant hazards consideration comments received: No.

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

Date of amendment request: March 18, 2004, as supplemented by letters dated April 7 and 13, 2004.

Brief description of amendments: The amendments revise TS Surveillance Requirement (SR) 4.7.7.e.3 to add a footnote that allows use of alternate criteria for those measured points at positive pressure but that do not meet the ⅛ inch Water Gauge criterion of the current TS. In addition the word ''that'' in the second line of the original text of SR 4.7.7.e.3 is changed to ''than'' to correct an existing typographical error. These amendments supersede Notice of Enforcement Discretion (NOED) No. 04–6–001, which the Commission staff granted to STPNOC on March 23, 2004.

Date of issuance: April 15, 2004.

Effective date: As of the date of issuance.

Amendment Nos.: Unit 1–161; Unit 2–151.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC):

Yes. A notice was published in the **Federal Register** on April 5, 2004 (69 FR 17718). The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided an opportunity to request a hearing within 60 days from the date of publication, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment. The supplements dated April 7 and 13, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 15, 2004.

Tennessee Valley Authority, Docket Nos. 50–259, 50–260, and 50–296, Browns Ferry Nuclear Plant, Units 1, 2, and 3, Limestone County, Alabama

Date of application for amendments: April 11, 2003, as supplemented by the October 2, 2003, meeting, and a letter dated February 20, 2004.

Description of amendment request: The amendments revised Technical Specification (TS) Table 3.3.5.1–1 which will result in a change to the Updated Final Safety Analysis Report (UFSAR), Table 6.5–3.

Date of issuance: April 1, 2004.

Effective date: Date of issuance, to be implemented within 60 days for Unit 1, during Cycle 13 Refueling Outage for Unit 2 , and during Cycle 12 Refueling Outage for Unit 3.

Amendment Nos.: 250, 289 & 248.

Facility Operating License Nos. DPR–33, DPR–52, and DPR–68: Amendments revised the TSs which will result in a change the UFSAR, Table 6.5–3.

*Date of initial notice in **Federal Register:*** May 27, 2003 (68 FR 28857). The October 2, 2003, meeting, and the February 20, 2004, letter, provided clarifying information that did not change the scope of the original request or the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 1, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 27, 2003, as supplemented by letters dated December 9, 2003, January 14 and April 5, 2004.

Brief description of amendment: The amendment approves the application of leak-before-break methodology for the accumulator and residual heat removal lines and installation of an opening in the secondary shield wall in terms of the effect of the opening on occupational exposure. The shield wall opening is related to plant modifications that would facilitate maintenance on the replacement steam generators to be installed in Refueling Outage 14 (Fall 2005).

Date of issuance: April 12, 2004.

Effective date: April 12, 2004, and shall be implemented prior to entering Mode 4 during the startup from Refueling Outage 13 which is scheduled for the Spring of 2004.

Amendment No.: 161.

Facility Operating License No. NPF–30: The amendment revised the Final Safety Analysis Report.

*Date of initial notice in **Federal Register:*** July 22, 2003 (68 FR 43397). The December 9, 2003, January 14 and April 5, 2004, supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 12, 2004.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket No. 50–339, North Anna Power Station, Unit 2, Louisa County, Virginia

Date of application for amendment: March 28, 2002, as supplemented by letters dated May 13, June 19, July 9, July 25, August 2, August 16, and November 15, 2002, May 6, May 9, May 27, June 11 (2 letters), July 18, August 20, August 26, September 4, September 5, September 22, September 26 (2 letters), November 10, December 8, and December 17, 2003, and January 6, January 22 (2 letters), February 12, February 13, and March 1, 2004. The November 15, 2002, submittal replaced the submittals dated July 9, July 25, and August 16, 2002.

Brief description of amendment: This amendment revises Improved Technical Specification Sections 2.1, 4.2, and 5.6.5 in order to allow Virginia Electric and Power Company to implement Framatome ANP Advanced Mark-BW fuel at North Anna Power Station, Unit 2.

Date of issuance: April 1, 2004.

Effective date: As of the date of issuance and shall be implemented prior to the initiation of core onload during Refueling Outage 16 (Spring 2004).

Amendment No.: 216.

Renewed Facility Operating License No. NPF–7: Amendment changes the Improved Technical Specifications.

*Date of initial notice in **Federal Register:*** July 22, 2003 (68 FR 43397). The supplements dated July 18, August 20, August 26, September 4, September 5, September 22, September 26 (2 letters), November 10, December 8, and December 17, 2003, and January 6, January 22 (2 letters), February 12, February 13, and March 1, 2004, contained clarifying information only and did not change the initial no significant hazards consideration determination or expand the scope of the initial application.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 1, 2004.

No significant hazards consideration comments received: No.

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

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

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

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

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

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

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

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22.

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

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

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

800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov*. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

Pacific Gas and Electric Company, Docket No. 50–275, Diablo Canyon Power Plant, Unit No. 1, San Luis Obispo County, California

Date of application for amendment: April 2, 2004, as superseded by application dated April 8, 2004.

Description of amendment request: The amendment revises the Technical Specification 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation," to allow performance of Surveillance Requirement (SR) 3.3.5.2 for the trip actuation device operational test, prior to first entry into MODE 4, by adding a note to the FREQUENCY column of SR 3.3.5.2 on a one-time basis.

Date of issuance: April 15, 2004.

Effective date: April 15, 2004, and shall be implemented within 10 days from the date of issuance.

Amendment No.: 165.

Facility Operating License No. DPR–80: The amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. A public notice was published in the San Luis Obispo Tribune on April 13 and 14, 2004. The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received.

The Commission's related evaluation of the amendment, finding of exigent circumstances, state consultation, and

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

final NSHC determination are contained in a safety evaluation dated April 15, 2004.

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

NRC Section Chief: Stephen Dembek.

Dated in Rockville, Maryland, this 19th day of April, 2004.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

Director, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 04–9225 Filed 4–26–04; 8:45 am]

BILLING CODE 7590–01–P

PENSION BENEFIT GUARANTY CORPORATION

Proposed Submission of Information Collection for OMB Review; Comment Request; Termination of Single-Employer Plans, Missing Participants

AGENCY: Pension Benefit Guaranty Corporation.

ACTION: Notice of intention to request extension of OMB approval.

SUMMARY: The Pension Benefit Guaranty Corporation intends to request that the Office of Management and Budget (''OMB'') extend approval (with modifications), under the Paperwork Reduction Act of 1995, of a collection of information in its regulations on Termination of Single-Employer Plans and Missing Participants, and implementing forms and instructions (OMB control number 1212–0036; expires August 31, 2004). This notice informs the public of the PBGC's intent and solicits public comment on the collection of information.

DATES: Comments should be submitted by June 28, 2004.

ADDRESSES: Comments may be mailed to the Office of the General Counsel, Pension Benefit Guaranty Corporation, 1200 K Street, NW., Washington, DC 20005–4026, or delivered to Suite 340 at that address during normal business hours. Comments also may be submitted electronically through the PBGC's Web site at *www.pbgc.gov/paperwork*, or by fax to (202) 326–4112. The PBGC will make all comments available on its Web site, *www.pbgc.gov*.

Copies of the collection of information may be obtained without charge by writing to the PBGC's Communications and Public Affairs Department at Suite 240 at the above address or by visiting that office or calling (202) 326–4040 during normal business hours. (TTY and TDD users may call the Federal relay service toll-free at 1–800–877–8339 and ask to be connected to (202) 326–4040.) The regulations and forms and instructions relating to this collection of information may be accessed on the PBGC's Web site at *www.pbgc.gov*.

FOR FURTHER INFORMATION CONTACT: Catherine B. Klion, Attorney, Office of the General Counsel, PBGC, 1200 K Street, NW., Washington, DC 20005–4026; (202) 326–4024. (TTY and TDD users may call the Federal relay service toll-free at 1–800–877–8339 and ask to be connected to (202) 326–4024.)

SUPPLEMENTARY INFORMATION: Under section 4041 of the Employee Retirement Income Security Act of 1974, as amended, a single-employer pension plan may terminate voluntarily only if it satisfies the requirements for either a standard or a distress termination. Pursuant to ERISA section 4041(b), for standard terminations, and section 4041(c), for distress terminations, and the PBGC's termination regulation (29 CFR part 4041), a plan administrator wishing to terminate a plan is required to submit specified information to the PBGC in support of the proposed termination and to provide specified information regarding the proposed termination to third parties (participants, beneficiaries, alternate payees, and employee organizations). In the case of a plan with participants or beneficiaries who cannot be located when their benefits are to be distributed, the plan administrator is subject to the requirements of ERISA section 4050 and the PBGC's missing participants regulation (29 CFR part 4050). The PBGC is making clarifying, simplifying, editorial, and other changes to the existing forms and instructions.

The PBGC estimates that 1,175 plan administrators will be subject to the collection of information requirements in the PBGC's termination and missing participants regulations and implementing forms and instructions each year, and that the total annual burden of complying with these requirements is 1,743 hours and $1,973,075. (Much of the work associated with terminating a plan is performed for purposes other than meeting these requirements.)

Comments on these collection of information requirements may address (among other things)—

- Whether the collection of information is necessary for the proper performance of the functions of the PBGC, including whether the information will have practical utility;
- The accuracy of the PBGC's estimate of the burden of the proposed collection of information, including the validity of the methodology and assumptions used;
- Enhancing the quality, utility, and clarity of the information to be collected; and
- Minimizing the burden of the collection of information on those who are to respond, including through the use of appropriate automated, electronic, mechanical, or other technological collection techniques or other forms of information technology, *e.g.*, permitting electronic submission of responses.

Issued in Washington, DC, this 21st day of April, 2004.

Stuart A. Sirkin,

Director, Corporate Policy and Research Department, Pension Benefit Guaranty Corporation.

[FR Doc. 04–9529 Filed 4–26–04; 8:45 am]

BILLING CODE 7708–01–P

RAILROAD RETIREMENT BOARD

Privacy Act of 1974; Proposed Changes to System of Records

AGENCY: Railroad Retirement Board (RRB).

ACTION: Notice of a revision of a Privacy Act System of Records.

SUMMARY: The purpose of this document is to give notice of changes to several categories of information in RRB–42, Uncollectible Benefit Overpayment Accounts. The RRB proposes to expand the scope of the system to include employee salary overpayments. Currently the system includes only benefit payments.

DATES: The changes to this System of Records shall become effective as proposed without further notice in 40 calendar days from the date of this publication unless comments are received before this date that would result in further modifications.

ADDRESSES: Send comments to Beatrice Ezerski, Secretary to the Board, Railroad Retirement Board, 844 N. Rush St., Chicago, Illinois 60611–2092.

FOR FURTHER INFORMATION CONTACT: LeRoy Blommaert, Privacy Act Officer, Railroad Retirement Board, 844 N. Rush St., Chicago, Illinois 60611–2092, telephone number (312) 751–4548, e-mail address, *blommlf@rrb.gov*.

SUPPLEMENTARY INFORMATION: The RRB proposes to expand the scope of the system to include employee salary overpayments. Currently the system includes only benefit overpayments.