STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of August 13, 2007

Tuesday, August 14, 2007.

9:30 a.m. Discussion of Intragovernmental Affairs (Closed-Ex. 1 & 9).

Week of August 20, 2007—Tentative

Tuesday, August 21, 2007.

1:25 p.m.

Affirmation Session (Public Meeting) (Tentative).

a. Final E-Filing Rule (Tentative).

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

1:30 p.m.

Meeting with OAS and CRCPD (Public Meeting). (Contact: Shawn Smith, 301 415–2620).

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

Wednesday, August 22, 2007.

9:30 a.m.

Periodic Briefing on New Reactor Issues (Morning Session)(Public Meeting) (Contact: Donna Williams, 301 415–1322).

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

1:30 p.m.

Periodic Briefing on New Reactor Issues (Afternoon Session)(Public Meeting) (Contact: Donna Williams, 301 415–1322).

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

Week of August 27, 2007—Tentative

There are no meetings scheduled for the Week of August 27, 2007.

Week of September 3, 2007—Tentative

There are no meetings scheduled for the Week of September 3, 2007.

Week of September 10, 2007—Tentative

There are no meetings scheduled for the Week of September 10, 2007.

Week of September 17, 2007—Tentative

There are no meetings scheduled for the Week of September 17, 2007.

*The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: Michelle Schroll, (301) 415–1662.

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

The NRC provides reasonable accommodation to individuals with disabilities where appropriate. If you need a reasonable accommodation to participate in these public meetings, or need this meeting notice or the transcript or other information from the public meetings in another format (e.g. braille, large print), please notify the NRC's Disability Program Coordinator, Rohn Brown, at 301–492–2279, TDD: 301–415–2100, or by e-mail at *REB3@nrc.gov.* Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to *dkw@nrc.gov.*

Dated: August 9, 2007.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 07–3987 Filed 8–10–07; 11:37 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from July 19, 2007, to August 1, 2007. The last biweekly notice was published on July 31, 2007 (72 FR 41780).

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Date of amendments request: June 29, 2007.

Description of amendments request: The amendment would modify Technical Specification (TS) requirements related to control room envelope (CRE) habitability in TS 3.7.8, ''Control Room Emergency Ventilation System (CREVS),'' and TS 5.5, ''Programs and Manuals.'' The changes are consistent with the Nuclear Regulatory Commission approved Technical Specification Task Force (TSTF)–448, Revision 3, ''Control Room Habitability.'' The availability of the TS improvement was published in the **Federal Register** on January 17, 2007 (72 FR 2022) as part of the consolidated item improvement process (CLIIP). In addition, the amendment would remove a footnote currently contained in the Completion Time of TS 3.7.8, Required Action D. The footnote was added in Amendment Nos. 250/227 and was only applicable during the Unit 1 2002 refueling outage.

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

The removal of a footnote [to TS 3.7.8] that is no longer applicable is an editorial change that does not affect accident initiators or precursors, nor alter the design assumptions, conditions or configuration of the facility. The proposed change also does not affect the ability of SSCs to perform their intended function to mitigate the consequences of an accident. Therefore, the proposed editorial change does not increase the probability or consequences of an accident 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 does not impact the accident analysis. The proposed change does not alter the required mitigation capability of the CRE emergency ventilation system, or its functioning during accident conditions as assumed in the licensing basis analyses of design basis accident radiological consequences to CRE occupants. No new or different accidents result from performing the new surveillance or following the new program. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a significant change in the methods governing normal plant operation. The proposed change does not alter any safety analysis assumptions and is consistent with current plant operating practice.

The proposed change is the editorial removal of a footnote [to TS 3.7.8] that no longer applies. The removal of a footnote that no longer applies does not impact the accident analyses. Additionally, it does not add or modify any existing plant equipment and does not introduce any new operational methods. Therefore, the proposed editorial change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

Response: No.

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

The proposed editorial change [removal of a footnote to TS 3.7.8] does not affect safety analyses acceptance criteria or safety system operation. Removal of a footnote that is no longer applicable does not result in plant operation outside the design basis. Therefore, the proposed editorial change does not involve a reduction in the margin of safety.

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

NRC Branch Chief: Mark G. Kowal.

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

Date of amendment request: February 20, 2007.

Description of amendment request: The proposed change would revise Limerick Generating Station (LGS), Units 1 and 2, Technical Specifications (TSs), Section 6.8.4.g, ''Primary Containment Leakage Rate Testing Program,'' to allow a one-time extension of no more than 5 years for the Type A, Integrated Leakage Rate Test (ILRT) interval. This revision is a one-time exception to the 10-year frequency of the performance-based leakage rate testing program for Type A tests as defined in Nuclear Energy Institute (NEI) document NEI 94–01, Revision 0, ''Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,'' pursuant to Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix J, Option B. The requested exception is to allow the ILRT to be performed within 15 years from the last ILRT as opposed to the current 10-year frequency. The most recent containment Type A ILRTs for LGS Units 1 and 2 were performed on May 15, 1998, and May 21, 1999, respectively.

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

1. Does the proposed amendment involve a significant increase in the probability or

consequences of an accident previously evaluated?

Response: No.

The proposed change will revise TS 6.8.4.g (''Primary Containment Leakage Rate Testing Program'') of the LGS, Units 1 and 2 TS to reflect a one-time extension to the Type A Integrated Leak Rate Test (ILRT) as currently specified in the Technical Specifications. This change will extend the requirement to perform the Type A ILRT from the current requirement of 10 to 15 years, which is ''no later than May 15, 2013'' for LGS, Unit 1 and is ''no later than May 21, 2014'' for Unit 2.

The function of the containment is to isolate and contain fission products released from the reactor coolant system following a design basis Loss of Coolant Accident (LOCA) and to confine the postulated release of radioactive material to within limits. The test interval associated with Type A ILRTs is not a precursor of any accident previously evaluated. Type A ILRTs provide assurance that the LGS, Units 1 and 2 containments will not exceed allowable leakage rate values specified in the TS and will continue to perform their design function following an accident. The risk assessment of the proposed change has concluded that there is an insignificant increase in Large Early Release Frequency, Person-Rem, and Conditional Containment Failure Frequency. Additionally, containment inspections have also been performed which demonstrate the continued structural integrity of the primary containment and will be performed in the future as required by the ASME Code.

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

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

Response: No.

The proposed change for a one-time extension of the Type A ILRTs for LGS, Units 1 and 2 will not affect the control parameters governing unit operation or the response of plant equipment to transient and accident conditions. The proposed change does not introduce any new equipment, modes of system operation or failure mechanisms.

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 amendment involve a significant reduction in a margin of safety?

Response: No.

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

EGC has conducted a risk assessment to determine the impact of a change to the LGS, Units 1 and 2 Type A ILRT from 10 to 15 years. This risk assessment measured the impact to the Large Early Release Frequency, Person-Rem, and Conditional Containment Failure Frequency. This assessment indicated that the proposed LGS, Units 1 and 2 Type A ILRT interval extension has a very small change in risk to the public and is an acceptable plant change from a risk perspective.

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

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

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

NRC Branch Chief: Harold K. Chernoff.

Florida Power and Light Company, Docket Nos. 50–335 and 50–389, St. Lucie Plant, Units 1 and 2, St. Lucie County, Florida

Date of amendment request: June 4, 2007.

Description of amendment request: The proposed amendment would remove the Technical Specification (TS) requirements that reference hydrogen recombiners and hydrogen monitors. The proposed amendment suggests changes support implementation of the revisions to 10 CFR 50.44, ''Standards for Combustible Gas Control System in Light Water Cooled Power Reactors,'' that became effective on September 16, 2003. The changes would be consistent with Revision 1 of the NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–447, ''Elimination of Hydrogen Recombiners and Change to Hydrogen and Oxygen Monitors.'' The particular TS improvement in question was announced in the **Federal Register** Notice on September 25, 2003, as part of the consolidated line item improvement process.

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

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 the 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 the 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.

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Branch Chief: Thomas H. Boyce.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of amendment request: June 13, 2007.

Description of amendment request: The proposed amendment would revise Technical Specifications (TS) 5.5.9, ''Ventilation Filter Testing Program (VFTP),'' to impose lower (i.e., more restrictive) limits on the maximum pressure drop across the combined high efficiency particulate air filters and charcoal adsorbers in three safety-related ventilation systems. These ventilation systems are the Control Room Emergency Ventilation System, the Engineered Safety Features Ventilation System, and the Fuel-Handling Area Exhaust Ventilation System.

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

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

Response: No.

The proposed change consists of establishing more restrictive criteria in the Technical Specification (TS) for the maximum pressure drop across high efficiency particulate air filters (HEPA) and charcoal adsorbers in safety-related ventilation systems. These TS criteria are used to determine the acceptability of periodic test results. These criteria are not accident initiators. Therefore, there will be no effect on the probability of an accident. The safety-related ventilation systems involved in the proposed change function to mitigate the consequences of accidents. The proposed change will provide increased assurance that the HEPA filters and charcoal adsorbers in these systems will be capable of performing their safety function of reducing the release of radioactive material resulting from evaluated accidents. Therefore, there will be no increase in the consequences of those accidents.

Therefore, the proposed change will 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 consists of establishing more restrictive acceptance criteria for existing TS[-]required tests. The proposed change does not affect the manner in which the tests are performed. The proposed change will not result in any new or different methods or modes of operation of existing structures, systems, or components. The proposed change will not introduce any new structures, system, or components.

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

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

Response: No.

The margin of safety associated with the proposed change is the capability of the applicable safety-related ventilation systems to prevent radiation exposures from exceeding acceptable limits due to the release of radioactive material caused by an evaluated accident. The proposed change will provide increased assurance that the HEPA filters and charcoal adsorbers in these systems will be capable of performing this function.

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

The Nuclear Regulatory Commission (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: Kimberly Harshaw, Esquire, One Cook Place, Bridgman, MI 49106.

NRC Acting Branch Chief: Travis Tate.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of amendment request: June 27, 2007.

Description of amendment request: The proposed amendment would revise Technical Specifications (TS) Surveillance Requirements 3.8.1.2, 8, 12, 13, 16, and 19, changing the steady state frequency of all diesel generators (DGs) from the current allowed frequency range of 59.4–61.2 Hz, to 59.4–60.5 Hz (i.e., a decrease of the upper limit, resulting in narrowing of the current range). The licensee stated that the current frequency range is nonconservative and could result in undesirable effects such as centrifugal charging pump motor brake horsepower exceeding its nameplate maximum horsepower, and overloading the DGs.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee provided its analysis of the issue of no significant hazards consideration. The NRC staff has performed its own analysis, 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 more restrictive steady state frequency range ensures that the diesel generators and equipment being powered by the diesel generators will function as designed to mitigate an accident as described in the Update Final Safety Analysis Report (UFSAR). The DGs and the equipment they power are part of the systems required to mitigate accidents; no accident analyzed in the UFSAR is initiated by mitigation equipment. Therefore, the proposed change to the allowed frequency range of the DGs will not have any impact on the probability of an accident previously evaluated. Furthermore, other than narrowing the allowed frequency range of the DGs, there is no other design or operational change. Therefore, the proposed change does not increase the probability of malfunction of the DGs or the equipment they power.

Narrowing of the DG maximum steady state frequency limit will ensure that the DGs and equipment powered by the DGs will perform as originally designed and analyzed to mitigate the consequences of any accident described in the UFSAR. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated in the UFSAR.

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

Response: No.

There is no design change associated with the proposed amendment. Making an existing DG requirement more restrictive alone will not alter plant configuration because no new or different type of equipment will be installed, and because no methods governing plant operation will be changed. The proposed change to allowed frequency range will not have any effect on the assumptions of accident scenarios previously made in the UFSAR. 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.

Despite the proposed change to the DG maximum steady state frequency limit, the DGs and equipment powered by the DGs will continue to perform as originally designed, and originally analyzed in the UFSAR. There is no associated change to the methods and assumptions used to analyze DG performance. The proposed change will maintain the required function of the DGs and the equipment powered by the DGs to ensure that operation of structures, systems, or components is as currently set forth in the UFSAR. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Kimberly Harshaw, Esquire, One Cook Place, Bridgman, MI 49106.

NRC Acting Branch Chief: Travis L. Tate.

Nuclear Management Company, LLC, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of amendment request: July 9, 2007.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS), an NRC-controlled document, by moving the Table of Contents (TOC) out of the TS and making the TOC into a licensee-controlled document.

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the Code of Federal Regulations (10 CFR) Part 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration (NSHC) which is reproduced below:

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

No.

The proposed change is administrative and affects control of a document, the TOC, listing the specifications in the plant TS. Transferring control from the NRC to NMC (the licensee) does not affect the operation, physical configuration, or function of plant equipment or systems. It does not impact the initiators or assumptions of analyzed events, nor does it impact the mitigation of accidents or transient events. The change has no impact on, and hence cannot increase, the probability or consequences of an accident previously evaluated.

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

No.

The proposed change is administrative and does not alter the plant configuration, require installation of new equipment, alter assumptions about previously analyzed accidents, or impact the operation or function of plant equipment or systems. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No.

The proposed change is administrative. The TOC is not required by regulation to be in the TS. [Its] removal does not impact any safety assumptions or have the potential to reduce a margin of safety as described in the TS Bases. The change involves a transfer of control of the TOC from the NRC to NMC. No change in the technical content of the TS [ ] is involved. Consequently, transfer from the NRC to NMC has no impact on the margin of safety, and hence cannot involve a significant reduction in the margin of safety.

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

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

NRC Acting Branch Chief: Travis L. Tate.

PPL Susquehanna, LLC, Docket No. 50–388, Susquehanna Steam Electric Station, Unit 2

(SSES 2), Luzerne County, Pennsylvania.

Date of amendment request: March 2, 2007.

Description of amendment request: The proposed amendment would add an ACTIONS Note 3 to the SSES 2 Technical Specification 3.8.1, ''AC Sources—Operating,'' to allow a Unit 1 4160 volt subsystem to be de-energized and removed from service to perform bus maintenance.

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

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

Response: No.

This change does not involve any physical change to structures, systems, or components (SSCs) and does not alter the method of operation of any SSCs. The current assumptions in the safety analysis regarding accident initiators and mitigation of accidents are unaffected by these changes. No

SSC failure modes or mechanisms are being introduced, and the likelihood of previously analyzed failures remains unchanged.

Operation in accordance with the proposed new ACTIONS Note 3 in Unit 2 Technical Specification 3.8.1 ensures that the AC [alternating current] distribution system and supported equipment remain capable of performing their functions as described in the Final Safety Analysis Report (FSAR). There are no changes to any accident initiators or to the mitigating capability of safety-related equipment supported by the Class 1E Electrical AC system. The protection provided by these safety-related systems will continue to be provided as assumed by the safety analysis.

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

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

Response: No.

The proposed change does not involve a physical alteration of any plant equipment. 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 will the function demands on credited equipment be 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 [final safety analysis report]. As such, no new failure modes are being introduced. The change does not alter assumptions made in the safety analysis and licensing basis.

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

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

Response: No.

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed change is acceptable because the new ACTIONS Note 3 has been established to be consistent with the existing completion times for declaring required equipment inoperable that has no offsite power or DG [diesel generator] power available. Therefore, the plant response to analyzed events is not affected by this change and will continue to provide the margin of safety assumed by the safety analysis.

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

NRC Branch Chief: Mark G. Kowal.

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

Date of amendment request: July 26, 2007

Description of amendment request: The proposed amendment would remove values for turbine first stage pressure equivalent to P_{bypass} from the Technical Specifications. P_{bypass} is the reactor power level below which the turbine stop valve closure and the turbine control valve fast closure reactor protection system trip functions and the end-of-cycle recirculation pump trip are bypassed automatically.

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

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

Response: No.

The proposed removal of values for turbine first stage pressure associated with P_{bypass} from the Technical Specifications does not alter the requirements for component operability or surveillance currently in the Technical Specifications. The proposed change will have no impact on any safety related structures, systems or components.

The probability of occurrence of a previously evaluated accident is not increased because this change does not introduce any new potential accident initiating conditions. The consequences of accidents previously evaluated in the UFSAR [Updated Final Safety Analysis Report] are not affected because the ability of the components to perform their required function is not affected.

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

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

Response: No.

The proposed change is administrative in nature, and does not result in physical alterations or changes in the method by which any safety related system performs its intended function. The proposed change does not affect any safety analysis assumptions. The proposed change does not create any new accident initiators or involve an activity that could be an initiator of an accident of a different type.

All components will continue to be tested to the same requirements as defined in the Technical Specification Surveillance Requirements. The proposed revision does not make changes in any method of testing or how any safety related system performs its safety functions.

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

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

Response: No.

The proposed change to remove values for turbine first stage pressure associated with P_{bypass} from the Technical Specifications does not alter the Technical Specification requirements for reactor protection system operability. The turbine first stage pressure setpoint will be controlled in accordance with plant procedures and will be verified during post-installation testing.

The proposed change will not affect the current Technical Specification requirements or the components to which they apply.

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

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

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

NRC Branch Chief: Harold K. Chernoff.

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

Date of amendment request: June 26, 2007.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TSs) to establish more effective and appropriate action, surveillance, and administrative requirements related to ensuring the habitability of the control room envelope (CRE) in accordance with Nuclear Regulatory Commission (NRC)-approved TS Task Force (TSTF) Standard Technical Specification change traveler TSTF–448, Revision 3, ''Control Room Habitability.'' Specifically, the proposed amendment would modify TS 3.7.7, Control Room Makeup and Cleanup Filtration System (CRMCFS) and TS Section 6.8, ''Administrative Controls-Procedures, Programs, and Manuals.'' The NRC staff issued a ''Notice of Availability of Technical Specification Improvement to

Modify Requirements Regarding Control Room Envelope Habitability Using the Consolidated Line Item Improvement Process'' associated with TSTF–448, Revision 3, in the **Federal Register** on January 17, 2007 (72 FR 2022). The notice included a model safety evaluation, a model no significant hazards consideration (NSHC) determination, and a model license amendment request. In its application dated June 26, 2007, the licensee affirmed the applicability of the model NSHC determination which is presented below.

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

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

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

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

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

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

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

The NRC staff has reviewed the analysis adopted by the licensee and, based on this review, it appears that the standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves NSHC.

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

NRC Branch Chief: Thomas G. Hiltz.

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

Date of amendment request: April 10, 2007.

Brief description of amendments: The proposed amendments would revise Technical Specifications (TS) 3.1, ''Reactivity Control Systems,'' TS 3.2, ''Power Distribution Limits,'' TS 3.3, ''Instrumentation,'' and TS 5.6.5b, ''Core Operating Limits Report (COLR).'' The requested change proposes to incorporate standard Westinghouse-developed and NRC-approved analytical methods into the lists of methodologies used to establish the core operating limits.

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

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

Response: No.

No physical plant changes or changes in manner in which the plant will be operated as a result of the methodology changes. The proposed changes do not impact the condition or performance of any plant structure, system or component. The core operating limits are established to support Technical Specifications 3.1, 3.2, 3.3, and 3.4. The core operating limits ensure that fuel design limits are not exceeded during any conditions of normal operation or in the event of any Anticipated Operational Occurrence (AOO). The methods used to establish the core operating limits for each operating cycle are based on methods previously found acceptable by the NRC and listed in Technical Specifications section 5.6.5.b. Application of these NRC-approved methods will continue to ensure that acceptable operating limits are established to protect the fuel cladding integrity during normal operation and AOOs. The requested Technical Specification changes, including those changes proposed to conform with the NRC-approved analysis methodologies, do not involve any plant modifications or operational changes that could affect system reliability, performance, or possibility of operator error. The requested changes do not affect any postulated accident precursors, does not affect any accident mitigation systems, and does not introduce any new accident initiation mechanisms.

As a result, the proposed changes to the CPSES [Comanche Peak Steam Electric Station] Technical Specifications do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated since neither accident probabilities nor consequences are being affected by this proposed change.

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

Response: No.

There are no physical changes being made to the plant. No new modes of plant operation are being introduced. The parameters assumed in the analyses are within the design limits of the existing plant equipment. All plant systems will perform as designed during the response to a potential accident.

Therefore, the proposed change to the CPSES Technical Specifications does not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any accident previously evaluated.

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

Response: No.

The NRC-approved accident analysis methodologies include restrictions on the choice of inputs, the degree of conservatism inherent in the calculations, and specified event acceptance criteria. Analyses performed in accordance with these methodologies will not result in adverse effects on the regulated margin of safety. Similarly, the use of axial power distribution controls based on the relaxed axial offset control strategy is a time-proven and NRC-approved method. The method is consistent with the accident analyses assumptions as described in the list of NRC-approved methodologies proposed to be used to establish the core operating limits. Finally, the proposed changes to allow operation with the BEACON [Best Estimate Analyzer for Core Operation Nuclear] power distribution monitoring tool provide additional information to the reactor operators on the state of the reactor core. Again, the use of the BEACON tool and the methodology used to develop the inputs to the tool are consistent with and controlled by the NRC-approved methodologies used to establish the core operating limits. As such, the margin of safety assumed in the plant safety analysis is not adversely affected by the proposed changes.

Based on the above evaluations, TXU Power concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of no significant hazards consideration is justified.

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

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

NRC Branch Chief: Thomas G. Hiltz.

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

Date of amendment request: May 22, 2007.

Brief description of amendments: The proposed amendment would revise the Technical Requirements Surveillance (TRS) 13.3.33.2, Cycling Frequency for the Turbine Stop and Control Valves. The proposed change would increase the frequency interval for the turbine stop and control valves testing from 12 to 26 weeks.

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

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

Response: No.

The proposed change will increase the frequency interval for testing the high pressure (HP) and low pressure (LP) turbine stop and control valves to 26 weeks. This test requires the movement of the HP and LP turbine valves through one complete cycle once every 26 weeks. The test verifies freedom of movement of the valve components and is beneficial in early detection of problems with valve operation. [The test ensures that all turbine steam inlet valves are capable of closing to protect the turbine from excessive overspeed, which could generate potentially damaging missiles.]

Siemens, the turbine manufacturer for Comanche Peak Steam Electric Station (CPSES), has evaluated the change in the probability of generating external/high-trajectory turbine missiles resulting from a hypothetical LP turbine disk failure which could adversely affect safety-related SSCs [structures, systems, and components] due to the change in the surveillance interval weeks using a previously approved missile probability analysis methodology. The results of the analysis show the new valve test interval of 26 weeks with a turbine inspection interval of 100,000 hours is safe and acceptable as the probability of occurrence of a turbine missile per turbine year is less than the Nuclear Regulatory Commission (NRC) limit of 1E–4 per 8760 hours (turbine year) or 11.42E–4 at 100,000 hours (Reference 7.4 [of the licensee's May 22, 2007, application]). Therefore, the risk of the loss of an essential system from a single event is acceptable. Since the probability of generating external, high-trajectory turbine missiles resulting from a hypothetical LP turbine disc failure which could adversely affect safety related SSCs due to the increased valve test interval from 12 to 26 weeks is less than the NRC limit, it is acceptable to increase the turbine test interval in TRS 13.3.33.2. The test interval change would increase overall plant capacity factor and result in a net improvement in plant safety by reducing the likelihood of plant trips and stress and wear on plant components. In addition, the increased test intervals would reduce the likely cause of a plant transient and unnecessary burden on personnel resources which is consistent with Generic Letter 93–005 (Reference 7.7 [of the licensee's May 22, 2007, application]) and NUREG–1366 (Reference 7.2 [of the licensee's May 22, 2007, application]). Based upon Siemens' analysis and the updated stop and control valves failure probability, it is concluded that the implementation of this change in testing frequency will not increase the probability or consequences of an accident previously evaluated in the UFSAR.

The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed change is consistent with safety analysis assumptions and resultant consequences.

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

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

Response: No.

The proposed change will reduce the frequency for testing the high pressure (HP) and low pressure (LP) turbine stop and control valves. Turbine overspeed is limited by rapid closure of the turbine stop and control valves. Turbine overspeed can result in the occurrence of turbine missiles from a burst type failure of the low pressure blades or disks. The damage from turbine missiles has been previously evaluated in the UFSAR [updated final safety analysis report] (Reference 7.3 [of the licensee's May 22, 2007, application]). The proposed activity does not introduce the possibility of a new accident because no new failure modes are introduced.

Turbine overspeed with the resulting turbine missiles is the only accident potentially affected by failure of the turbine stop and control valves. The turbine missile analysis is not altered by reducing the frequency of high and low pressure stop and control valve testing. Reducing the frequency of turbine valve testing from every 12 weeks to every 26 weeks does not result in a significant change in the failure rate, nor does it affect the failure modes for the turbine valves.

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. This amendment will not affect the normal method of plant operation or change any operating parameters. No performance requirements or response time limits will be affected. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this amendment. There will be no adverse effect or challenges imposed on any safety-related system as a result of this amendment.

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

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

Response: No.

The proposed change does not involve a significant reduction in a margin of safety since the conclusions of the safety analyses in the CPSES FSAR [final safety analysis report] (Reference 7.3 [of the licensee's May 22, 2007, application]) are essentially unchanged and NRC safety limits are not exceeded.

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

Based on the above evaluations, TXU Power concludes that the proposed amendment(s) present no significant hazards under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of ''no significant hazards consideration'' is justified.

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

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

NRC Branch Chief: Thomas G. Hiltz.

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

Date of amendment request: July 13, 2007.

Description of amendment request: The proposed amendment revises the Technical Specifications (TSs) requirements related to main control room and emergency switchgear room envelope habitability. These changes are consistent with the Nuclear Regulatory Commission (NRC)-approved Revision 3 of Technical Specification Task Force (TSTF) Standard Technical Specifications (STS) Change Traveler TSTF–448, ''Control Room Habitability.''

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

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes consist of TS wording, format and conforming changes to facilitate incorporation of TSTF–448 [72 FR 2022] into the Surry custom TS and for consistency with NUREG–1431, Revision 3, to the extent practical. The proposed changes are administrative in nature and, as such, do not impact the condition or performance of any plant structure, system or component. The proposed changes do not affect the initiators of any previously analyzed event or the assumed mitigation of accident or transient events. As a result, the proposed administrative changes to the Surry TS do not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated since neither accident probabilities or consequences are being affected by the proposed changes.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes are administrative in nature, and therefore do not involve any changes in station operation or physical modifications to the plant. In addition, no changes are being made in the methods used to respond to plant transients that have been previously analyzed. No changes are being made to plant parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions, and no new failure modes are being introduced. Therefore, the proposed changes to the Surry Technical Specifications do not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

3. Involve a significant reduction in a margin of safety.

The proposed changes consist of TS wording, format and conforming changes to facilitate incorporation of TSTF–448 into the Surry custom TS and for consistency with NUREG–1431, Revision 3. The proposed changes are administrative in nature, and do not impact station operation or any plant structure, system or component that is relied upon for accident mitigation. Furthermore, the margin of safety assumed in the plant safety analysis is not affected in any way by the proposed changes. Therefore, the proposed administrative changes to the Surry Technical Specifications do not involve a reduction in a margin of safety.

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Counsel, Dominion Resources Services, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385.

NRC Branch Chief: Evangelos C. Marinos.

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.

Exelon Generation Company, LLC, and PSEG Nuclear LLC,

Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3, York and Lancaster Counties, Pennsylvania

Date of amendment request: March 6, 2007.

Brief description of amendment request: The proposed amendment would modify the main steam isolation valve (MSIV) leakage Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.3.14 to establish a total leakage rate limit for the sum of the four main steam lines.

*Date of publication of individual notice in **Federal Register**:* July 24, 2007.

Expiration date of individual notice: September 22, 2007.

Tennessee Valley Authority, Docket No. 50–259, Browns Ferry Nuclear Plant, Unit 1, Limestone County, Alabama

Date of application for amendments: June 25, as supplemented July 3, 2007.

Description of amendments request: The proposed amendment would allow deletion of License Condition 2.(G)2 regarding the performance of power uprate large transient testing.

*Date of publication of individual notice in the **Federal Register**:* July 13, 2007 (72 FR 38627).

Expiration date of individual notice: August 14, 2007 (Public comments) and September 11, 2007 (Hearing requests).

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: September 28, 2006.

Brief description of amendment: The amendment revises the Oyster Creek Technical Specification (TS) definition of Channel Calibration, Channel Check, and Channel Test consistent with NUREG–1433, Revision 3.0, ''Standard Technical Specifications General Electric Plants, BWR/4 Specifications,'' dated June 2004. These definitions apply to all instrument functions in the TSs, including Reactor Protection System instruments.

Date of Issuance: July 27, 2007.

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

Amendment No.: 263.

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

Date of initial notice in ***Federal Register***: November 21, 2006 (71 FR 67392). The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated July 27, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 20, 2006, as supplemented by letter dated May 3, 2007.

Brief description of amendments: The amendments revised Technical Specifications (TS) 3.1.6, ''Shutdown Control Element Assembly (CEA) Insertion Limits,'' to modify the TS Limiting Condition for Operation (LCO) 3.1.6 and Surveillance Requirements (SRs) 3.1.6.1 to require shutdown CEAs to be withdrawn to ≥147.75 inches, instead of the current limit of ≥144.75 inches.

Date of issuance: July 25, 2007.

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

Amendment Nos.: Unit 1–168, Unit 2–168, Unit 3–168.

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

Date of initial notice in ***Federal Register***: September 26, 2006 (71 FR 56191). The supplement dated May 3, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 25, 2007.

No significant hazards consideration comments received: No.

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of application for amendment: February 2, 2007.

Brief description of amendment: This amendment deletes the technical specification (TS) requirements related to containment hydrogen monitors and supports implementation of the revisions of 10 CFR 50.44, Combustible Gas Control for Nuclear Power Reactors, that became effective on October 16, 2003. This is a Consolidated Line Item Improvement Program modification, which adopts TS Task Force (TSTF) Standard TS Change Traveler, TSTF–447, Elimination of Hydrogen Recombiners and Change to Hydrogen and Oxygen Monitors.

Date of issuance: July 16, 2007.

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

Amendment No.: 216.

Renewed Facility Operating License No. DPR–23: Amendment revises the technical specifications.

Date of initial notice in ***Federal Register***: April 24, 2007 (72 FR 20378). The Commission's related evaluation of the amendment is contained in a safety evaluation dated March 21, 2007.

No significant hazards consideration comments received: No.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of application for amendment: December 20, 2006.

Brief description of amendment: This amendment revises Technical Specification (TS) 6.12, ''High Radiation Area.'' The amendment aligns the requirements contained in the TS with the revised Regulatory Guide 8.38, Revision 1, ''Control of Access to High and Very High Radiation Areas in Nuclear Power Plants.'' Specifically, the changes include differentiating dose rates associated with high and very high radiation areas, adding requirements for groups entering high radiation areas, and clarifying the communication requirements for workers in high radiation areas.

Date of issuance: July 23, 2007.

Effective date: This amendment is effective as of the date of issuance and shall be implemented within 60 days of issuance.

Amendment No.: 125.

Facility Operating License No. NPF–63: Amendment revises the TSs.

Date of initial notice in ***Federal Register***: February 27, 2007 (72 FR 8802). The Commission's related evaluation of the amendment is contained in a safety evaluation dated July 23, 2007.

No significant hazards consideration comments received: No.

Duke Power Company LLC, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: July 31, 2006 as supplemented May 24, 2007.

Brief description of amendments: The amendments revised TS 3.6.3, ''Containment Isolation Valves,'' by removing the allowance to open the upper containment purge isolation valves in the applicable modes of operation when containment integrity is required by the TSs. In addition, the amendments deleted TS 3.3.6, ''Containment Purge and Exhaust Isolation Instrumentation''. The change made the TSs requirements consistent for both the upper and lower containment purge isolation valves.

Date of issuance: July 26, 2007.

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

Amendment Nos.: 243, 224.

Renewed Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the licenses and the technical specifications.

Date of initial notice in ***Federal Register***: December 5, 2006 (71 FR 70558) The supplement dated May 24, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 26, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 10, 2006, as supplemented by letter dated March 8, 2007.

Brief description of amendment: The changes would clarify technical specifications (TSs) for the Perry Nuclear Power Plant (PNPP) by revising the TS action requirements that must be followed when one or more annulus gas treatment system initiation channels are inoperable. The clarifying changes will make the PNPP TSs consistent with Nuclear Regulatory Commission (NRC) staff precedents for containment filtering safety systems that operate continuously in the protection mode of operation.

Date of issuance: July 30, 2007.

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

Amendment No.: 147.

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

Date of initial notice in ***Federal Register***: May 23, 2006 (71 FR 29678). The March 8, 2007, supplement contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 30, 2007.

No significant hazards consideration comments received: No.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant, Units 3 and 4, Miami-Dade County, Florida

Date of application for amendments: January 27, 2006, as supplemented November 28, 2006, April 30, 2007, and July 17, 2007.

Brief description of amendments: These amendments revise Technical Specifications (TS) Section 3/4 9.1, ''Boron Concentration,'' Section 3/4 9.14, ''Spent Fuel Storage,'' and Section 3/4 5.5.1, ''Fuel Storage Criticality'' to allow use of Metamic rack inserts, and administrative controls that require mixing higher reactivity fuel with lower-reactivity fuel.

Date of issuance: July 17, 2007.

Effective date: As of the date of issuance and shall be implemented prior to the end of Unit 4 Cycle 24.

Amendment Nos: 234 and 229.

Renewed Facility Operating License Nos. DPR–31 and DPR–41: Amendments revised the TS.

Date of initial notice in ***Federal Register***: May 9, 2006 (71 FR 26999). The supplements dated November 28, 2006, April 30, 2007, and July 17, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register** on May 9, 2006.

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

No significant hazards consideration comments received: No.

Nuclear Management Company, Docket No. 50–263, Monticello Nuclear Generating Plant (MNGP), Wright County, Minnesota

Date of application for amendment: January 29, 2007, as supplemented on June 5, 2007.

Brief description of amendment: The amendment revised Table 3.3.5.1–1 of the Technical Specifications for three low-pressure coolant injection loop select logic functions. The surveillance of these three functions was previously required to be performed every 92 days. The amended requirement requires a channel calibration and logic system functional test, respectively, every 24 months. In addition, the allowable values associated with these three functions are changed to match the extended surveillance interval.

Date of issuance: July 20, 2007.

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

Amendment No: 151.

Renewed Facility Operating License No. DPR–22: Amendment revised the Renewed Facility Operating License and Technical Specifications.

Date of initial notice in ***Federal Register***: March 13, 2007 (72 FR 11391).

The supplemental letter dated June 5, 2007, 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 July 20, 2007.

No significant hazards consideration comments received: No.

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

Date of amendment request: February 28, 2006, as supplemented by letters dated April 6 and May 31, 2007, and electronic mail dated July 18, 2007.

Brief description of amendments: The amendments revised TSs 3/4.8.2.1, ''DC [Direct Current] Sources—Operating,'' and 3/4.8.2.2, ''DC Sources—Shutdown,'' and add a new TS 3/4.8.2.3, ''Battery Parameters.'' The amendments revised allowed outage times for battery chargers as well as battery charger testing criteria, and relocate a number of battery surveillance requirements to a licensee-controlled Battery Monitoring and Maintenance Program. The changes are consistent with Standard TS Change Traveler TSTF–360, Revision 1, ''DC Electrical Rewrite.''

Date of issuance: July 20, 2007.

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

Amendment Nos.: Unit 1–180; Unit 2–167.

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

Date of initial notice in ***Federal Register***: September 12, 2006 (71 FR 53721). The supplemental letters dated April 6 and May 31, 2007, and electronic mail dated July 18, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 20, 2007.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 2nd day of August 2007.

For the Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. E7–15459 Filed 8–13–07; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Notice of Availability of the Final License Renewal Interim Staff Guidance LR–ISG–2006–03: Staff Guidance for Preparing Severe Accident Mitigation Alternatives Analyses

AGENCY: Nuclear Regulatory Commission (NRC).

ACTION: Notice of Availability.

SUMMARY: NRC is issuing its Final License Renewal Interim Staff Guidance LR–ISG–2006–03 for preparing severe accident mitigation alternatives (SAMA) analyses. This LR–ISG recommends that applicants for license renewal use the Guidance Document Nuclear Energy Institute 05–01, Revision A, (ADAMS Accession No. ML060530203) when preparing their SAMA analyses. The NRC staff issues LR–ISGs to facilitate timely implementation of the license renewal rule and to review activities associated with a license renewal application. The NRC staff will also incorporate the approved LR–ISG into the next revision of Supplement 1 to Regulatory Guide 4.2, ''Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses.''

ADDRESSES: The NRC maintains an Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. These documents may be accessed through the NRC's Public Electronic Reading Room on the Internet at *http://www.nrc.gov/reading-rm/adams.html.* Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS should contact the NRC Public Document Room (PDR) reference staff at 1–800–397–4209, 301–415–4737, or by e-mail at *pdr@nrc.gov.*

FOR FURTHER INFORMATION CONTACT: Mr. Richard L. Emch, Jr., Senior Project Manager, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001; telephone 301–415–1590 or by e-mail at *rle@nrc.gov.*

SUPPLEMENTARY INFORMATION: Attachment 1 to this **Federal Register** notice, entitled *Staff Position and Rationale for the Final License Renewal Interim Staff Guidance LR–ISG–2006–03: Staff Guidance for Preparing Severe Accident Mitigation Alternatives (SAMA)* Analyses contains the NRC staff's rationale for publishing the Final LR–ISG–2006–03. Attachment 2 to this **Federal Register** notice, entitled *Proposed License Renewal Interim Staff Guidance LR–ISG–2006–03: Staff Guidance for Preparing Severe Accident Mitigation Alternatives (SAMA) Analyses,* contains the guidance for preparing SAMA analyses related to license renewal applications. The NRC staff approves this LR–ISG for NRC and industry use. The NRC staff will also incorporate the approved LR–ISG into the next revision of Supplement 1 to Regulatory Guide 4.2, ''Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses.''

Dated at Rockville, Maryland, this 2nd day of August 2007.

For the Nuclear Regulatory Commission.

Pao-Tsin Kuo,

Director, Division of License Renewal, Office of Nuclear Reactor Regulation.

Attachment 1—Staff Position and Rationale for the Final License Renewal Interim Staff Guidance LR–ISG–2006–03: Staff Guidance for Preparing Severe Accident Mitigation Alternatives Analyses

Staff Position: The NRC staff recommends that applicants for license renewal follow the guidance provided in Nuclear Energy Institute (NEI) 05–01, ''Severe Accident Mitigation Alternatives (SAMA) Analysis—Guidance Document,'' Revision A, when preparing their SAMA analyses.

Rationale: The NEI developed a generic Guidance Document NEI 05–01, Revision A, to help clarify the NRC staff's expectations regarding the information that needs to be included in SAMA analyses. The NRC staff reviewed and concluded that NEI 05–01, Revision A, describes existing NRC regulations and facilitates complete preparation of SAMA analysis submittals. The staff finds that utilization of the guidance provided in NEI 05–01, Revision A, will result in improved quality in SAMA analyses and a reduction in the number of requests for additional information.

Attachment 2—Final License Renewal Interim Staff Guidance LR–ISG–2006–03: Staff Guidance for Preparing Severe Accident Mitigation Alternatives Analyses

Introduction

A severe accident mitigation alternatives (SAMA) analyses is required as part of a license renewal application, if a SAMA analysis has not already been performed for the plant and reviewed by the NRC staff. SAMA analyses have been performed and submitted to the NRC for all applications for license renewal received by the staff thus far. Therefore, this LR–ISG is being recommended as guidance consistent with our goal to more effectively and efficiently resolve license renewal issues identified by the staff or the industry.

Background and Discussion

After receiving extensive requests for additional information regarding the SAMA analyses, several applicants for license renewal concluded that they did not fully understand the kind of information that the NRC staff was expecting to see in SAMA analyses.

The Nuclear Energy Institute (NEI) developed a generic guidance document to help clarify the NRC staff's expectations regarding the information that should be submitted in SAMA analyses. On April 8, 2005, NEI submitted NEI 05–01, ''Severe Accident Mitigation Alternatives (SAMA) Analysis—Guidance Document.'' The NRC staff reviewed this guidance document, and by letter, dated July 12, 2005, provided comments on NEI 05–01. The NRC staff's comments were discussed during a public meeting between NEI and NRC on July 21, 2005.

On February 17, 2006, NEI submitted its NEI 05–01, Revision A, dated November 2005. The NRC staff reviewed and concluded that this version fully resolved the NRC staff's comments. In addition, the NRC staff concluded that NEI 05–01, Revision A, describes existing NRC regulations, and facilitates complete preparation of SAMA analysis submittals.

Some applicants for license renewal have submitted SAMA analyses using the guidance provided in NEI 05–01, Revision A. The NRC staff found improved quality in the submitted SAMA analyses and a reduction in the