same as for the first alternative and the proposed action. The burden, however, on the licensee, end-users, and regulators would be greater than that of the proposed action by requiring more frequent reporting by the licensee, requiring the end-users to appoint a person knowledgeable of pertinent regulations, requiring the end-users to leak test the units, and requiring the regulator to track the units.

5.0 Agencies and Persons Contacted

GE has distribution facilities located in Wilmington, MA, Newark, CA, and Lincolnton, NC. NRC contacted the radiation control programs of the States of Massachusetts, California, and North Carolina. These states had no objection to the proposed action in this EA.

NRC staff has determined that the proposed action will not affect listed species or critical habitat. Therefore, no further consultation is required under Section 7 of the Endangered Species Act. Likewise, NRC staff have determined that the proposed action is not the type of activity that has potential to cause effects on historic properties. Therefore, no further consultation is required under Section 106 of the National Historic Preservation Act.

6.0 Conclusion

The action that NRC is considering is to issue an amendment to License No. 20–23904–01E and an exemption from 10 CFR 32.26 to allow GE Field Service Engineers to service Entryscan explosives/narcotics walk-through detection devices at customer sites, and to allow GE to ship the Entryscan devices in parts for final assembly at customer sites. The NRC staff considered the environmental consequences of approving the license amendment and exemption, and has determined that the approval will have no adverse effect on public health and safety or the environment. Therefore, the NRC staff concludes that the proposed action is the preferred alternative, the environmental impacts associated with the proposed action do not warrant denial of the license amendment and exemption request.

7.0 Finding of No Significant Impact

The Commission has prepared this EA related to GE's exemption request. On the basis of this EA, the NRC finds that there are no significant environmental impacts from the proposed action, and that preparation of an environmental impact statement is not warranted. Accordingly, the NRC has determined that a Finding of No Significant Impact is appropriate.

8.0 References

1. SSD Certificate No. NR–0399–D–101–E.
2. NRC License No. 20–23904–01E.
3. GE letters dated November 29, 2006 and May 13, 2007, with enclosures thereto.

IV. Further Information

Questions regarding this action may be directed to Duncan White at (301) 415–2598 or by e-mail at *ADW@nrc.gov*.

Dated at Rockville, Maryland this 17th day of August, 2007.

For The Nuclear Regulatory Commission.

Janet Schlueter,

Director, Division of Materials Safety and State Agreements, Office of Federal and State Materials and Environmental Management Programs.

[FR Doc. E7–17878 Filed 9–10–07; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Federal Register Notice

AGENCY HOLDING THE MEETINGS: Nuclear Regulatory Commission.

DATES: Weeks of September 10, 17, 24, October 1, 8, 15, 2007.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of September 10, 2007

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.

Week of September 24, 2007—Tentative

There are no meetings scheduled for the Week of September 24, 2007.

Week of October 1, 2007—Tentative

Tuesday, October 2, 2007

9:30 a.m.
Periodic Briefing on Security Issues (Closed—Ex. 1 & 3).

Wednesday, October 3, 2007

2 p.m.
Briefing on NRC's International Programs, Performance, and Plans (Public Meeting) (Contact: Karen Henderson, 301–415–0202).

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

Week of October 8, 2007—Tentative

There are no meetings scheduled for the Week of October 8, 2007.

Week of October 15, 2007—Tentative

There are no meetings scheduled for the Week of October 15, 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: September 6, 2007.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 07–4468 Filed 9–7–07; 11:33 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 August 16, 2007 to August 29, 2007. The last biweekly notice was published on August 28, 2007 (72 FR 49568).

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 email to *pdr@nrc.gov*.

Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan

Date of amendment request: June 12, 2007.

Description of amendment request: The proposed amendment would revise Technical Specification 3.7.4 to add an Action Statement for two inoperable control center air conditioning (AC) subsystems. The proposed new Action Statement would allow a finite time to restore one control center AC subsystem to operable status and require verification that control room temperature remains < 90 °F every 4 hours.

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 by a reference to a generic analysis published in the **Federal Register** on December 18, 2006 (71 FR 75774), which is presented below:

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

The proposed change is described in Technical Specification Task Force (TSTF) Standard TS Change Traveler TSTF–477 adds an action statement for two inoperable control room subsystems.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). The proposed changes add an action statement for two inoperable control room subsystems.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). The proposed changes add an action statement for two inoperable control room subsystems. The equipment qualification temperature of the control room equipment is not affected. Future changes to the Bases or licensee-controlled document will be evaluated pursuant to the requirements of 10 CFR 50.59, ''Changes, test and experiments'', to ensure that such changes do not result in more than a minimal increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not adversely affect the ability of structures, systems and components (SSCs) to perform their intended safety function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological consequences of any accident previously evaluated. Further, the proposed changes do not increase the types and the amounts of radioactive effluent that may be released, nor significantly increase individual or cumulative occupation/public radiation exposures.

Therefore, the changes do not involve a significant increase in the probability or 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 proposed changes add an action statement for two inoperable control room subsystems. The changes do not involve a physical altering of the plant (i.e., no new or different type of equipment will be installed) or a change in methods governing normal plant operation. The requirements in the TS continue to require maintaining the control room temperature within the design limits.

Therefore, the changes do 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 proposed changes add an action statement for two inoperable control room subsystems. Instituting the proposed changes will continue to maintain the control room temperature within design limits. Changes to the Bases or license[e-] controlled document are performed in accordance with 10 CFR 50.59. This approach provides an effective level of regulatory control and ensures that the control room temperature will be maintained within design limits.

The proposed changes maintain sufficient controls to preserve the current margins of safety.

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

Attorney for licensee: David G. Pettinari, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Acting Branch Chief: Travis L. Tate.

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

Date of amendment request: July 17, 2007.

Description of amendment request: The proposed changes would modify Technical Specification (TS) requirements related to control room envelope (CRE) habitability in TS 3.7.3, ''Control Room Emergency Ventilation Air Supply (CREVAS) System'' and adds new TS 5.5.14, ''Control Room Envelope Habitability Program.''

These changes were proposed by the industry's TS Task Force (TSTF) and is designated TSTF–448. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on October 17, 2006 (71 FR 61075), on possible amendments concerning TSTF–448, including a model safety evaluation and model no significant hazards (NSHC) determination, using the consolidated line item improvement process (CLIIP). The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on January 17, 2007 (72 FR 2022). The licensee affirmed the applicability of the following NSHC determination in its application dated July 17, 2007.

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 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 a 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.

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

Attorney for licensee: Mr. William C. Dennis, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Branch Chief: Mark G. Kowal.

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

Date of amendment request: July 25, 2007.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TS) by adding an Action statement to the Limiting Condition for Operation (LCO) for TS 3.7.4, Control Room Air Conditioning (AC) System. The new Action statement allows a finite time to restore one control room AC subsystem to operable status (72 hours) and requires verification that control room temperature remains less than 104 °F every 4 hours. The licensing basis control room air temperature for the James A. FitzPatrick Nuclear Power Plant (JAFNPP) is 104 °F.

This change was proposed by the industry's TS Task Force (TSTF) and is designated TSTF–477. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on December 18, 2006 (71 FR 75774), on possible amendments concerning TSTF–477, including a model safety evaluation and model no significant hazards (NSHC) determination, using the consolidated line item improvement process (CLIIP). The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 26, 2007 (72 FR 14143). The licensee affirmed the applicability of the following NSHC determination in its application dated July 25, 2007.

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 Changes Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed change as described in Technical Specification Task Force (TSTF) Standard TS Change Traveler TSTF–477 adds an action statement for two inoperable control room subsystems.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). The proposed changes add an action statement for two inoperable control room subsystems. The equipment qualification temperature of the control room equipment is not affected. Future changes to the Bases or licensee controlled document will be evaluated pursuant to the requirements of 10 CFR 50.59, ''Changes, test and experiments,'' to ensure that such changes do not result in more than a minimal increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not adversely affect the ability of structures, systems and components (SSCs) to perform their intended safety function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological consequences of any accident previously evaluated. Further, the proposed changes do not increase the types and the amounts of radioactive effluent that may be released, nor significantly increase individual or cumulative occupation/public radiation exposures.

Therefore, the changes do not involve a significant increase in the probability or 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 Accident Previously Evaluated

The proposed changes add an action statement for two inoperable control room subsystems. The changes do not involve a physical altering of the plant (i.e., no new or different type of equipment will be installed) or a change in methods governing normal plant operation. The requirements in the TS continue to require maintaining the control room temperature within the design limits.

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

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

The proposed changes add an action statement for two inoperable control room subsystems. Instituting the proposed changes will continue to maintain the control room temperature within design limits. Changes to the Bases or license controlled document are performed in accordance with 10 CFR 50.59. This approach provides an effective level of regulatory control and ensures that the control room temperature will be maintained within design limits.

The proposed changes maintain sufficient controls to preserve the current margins of safety.

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

Attorney for licensee: Mr. William C. Dennis, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Branch Chief: Mark G. Kowal.

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

Date of amendment request: July 2, 2007.

Description of amendment request: The proposed amendment would modify RBS technical specification (TS) requirements for MODE change limitations in limiting condition for operation (LCO) 3.0.4 and surveillance requirement (SR) 3.0.4. The proposed TS changes are consistent with Revision 9 of Nuclear Regulatory Commission (NRC) approved Industry TS Task Force (TSTF) Standard TS Change Traveler, TSTF–359, ''Increase Flexibility in MODE Restraints.'' In addition, the proposed amendment would also change TS section 1.4, Frequency, Example 1.4–1, ''Surveillance Requirements,'' to accurately reflect the changes made by TSTF–359, which is consistent with NRC-approved TSTF–485, Revision 0, ''Correct Example 1.4–1.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 2, 2002 (67 FR 50475), as part of the Consolidated Line Item Improvement Process (CLIIP), on possible amendments to revise the plant-specific TS to modify requirements for MODE change limitations in LCO 3.0.4 and SR 3.0.4.

The NRC staff subsequently issued a notice of availability of the models for Safety Evaluation and No Significant Hazards Consideration Determination for referencing in license amendment applications in the **Federal Register** on April 4, 2003 (68 FR 16579). The licensee affirmed the applicability of the CLIIP, including the model No Significant Hazards Consideration Determination, in its application dated February 8, 2007.

The proposed TS changes are consistent with NRC-approved Industry TSTF Standard TS change, TSTF–359, Revision 8, as modified by 68 FR 16579. TSTF–359, Revision 8, was subsequently revised to incorporate the modifications discussed in the April 4, 2003, **Federal Register** notice and other minor changes. TSTF–359, Revision 9, was subsequently submitted to the NRC on April 28, 2003, and was approved by the NRC on May 9, 2003.

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

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

The proposed changes in TS Section 1.4, Frequency, Example 1.4–1, would accurately reflect the changes made by TSTF–359 in LCO 3.0.4 and SR 3.0.4, which are consistent with NRC-approved TSTF–485, Revision 0. These changes are considered administrative in that they modify the example to demonstrate the proper application of LCO 3.0.4 and SR 3.0.4. The requirements of LCO 3.0.4 and SR 3.0.4 are clear and are clearly explained in the associated Bases. As a result, modifying the example will not result in a change in usage of the TS.

The proposed changes in LCO 3.0.4 and SR 3.0.4 allow entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. The proposed changes do not adversely affect accident initiators or precursors, the ability of structures, systems, and components to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Being in a TS condition and the associated required actions are not an initiator of any accident previously evaluated. Therefore, the probability of an accident previously evaluated is not significantly increased. The consequences of an accident while relying on required actions as allowed by proposed LCO 3.0.4, are no different than the consequences of an accident while entering and relying on the required actions while starting in a condition of applicability of the TS. Therefore, the consequences of an accident previously evaluated are not significantly affected by these changes. The addition of a requirement to assess and manage the risk introduced by these changes will further minimize possible concerns. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No new or different accidents result from utilizing the proposed changes. The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The proposed changes do not alter assumptions made in the safety analysis and are consistent with the safety analysis assumptions and current plant operating practice. Entering into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by these changes will further minimize possible concerns. Thus, these changes do not create the possibility of a new or different kind of accident from an accident previously evaluated.

Criterion 3—The Proposed Changes Do Not Involve a Significant Reduction in the Margin of Safety

The proposed changes in TS section 1.4, Example 1.4–1, are considered administrative and will have no effect on the application of the TS requirements. Therefore, the margin of safety provided by the TS requirements is unchanged.

The proposed changes in TS LCO 3.0.4 and SR 3.0.4 allow entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. The RBS TS allows operation of the plant without the full complement of equipment through the TS conditions for not meeting the TS LCO. The risk associated with this allowance is managed by the imposition of required actions that must be performed within the prescribed completion times. The net effect of being in a TS LCO condition on the margin of safety is not considered significant. The proposed changes do not alter the required actions or completion times of the TS. The proposed changes allow TS conditions to be entered, and the associated required actions and completion times to be used in new circumstances. This use is predicated upon the licensee's performance of a risk assessment and the management of plant risk. The changes also eliminate current allowances for utilizing required actions and completion times in similar circumstances, without assessing and managing risk. The net change to the margin of safety is insignificant. Therefore, these changes do not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Thomas G. Hiltz.

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

Date of amendment request: July 16, 2007, as supplemented by letter dated August 7, 2007.

Description of amendment request: The proposed amendment would revise the facility operating license (FOL), Paragraph 2.C, and technical specifications (TS) 3.7.2 and TS 5.5.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on October 17, 2006 (71 FR 61075), on possible amendments to revise the plant-specific TS, to strengthen requirements regarding control room envelope (CRE) habitability by changing the action and surveillance requirements associated with the limiting condition for operability requirements for the CRE emergency ventilation system. A new TS administrative controls program on CRE habitability is being added, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line-item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on January 17, 2007 (72 FR 2022). The licensee affirmed the applicability of the model NSHC determination in its application dated July 16, 2007, as supplemented by letter dated August 7, 2007.

Basis for proposed NSHC 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 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 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 proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Branch Chief: Thomas G. Hiltz.

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

Date of amendment request: August 17, 2007.

Description of amendment request: The proposed amendment would revise the date for performing the ''Type A test'' in the RBS technical specification (TS) 5.5.13, ''Primary Containment Leak Rate Testing Program,'' from ''prior to December 14, 2007'' to ''April 14, 2008.''

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 amendment to TS 5.5.13 allows a one-time extension to the current interval for the ILRT [integrated leak rate test]. The current interval of 15 years 4 months, based on past performance, would be extended on a one-time basis to 15 years and 8 months from the date of the last test. The proposed extension to the ILRT cannot increase the probability of an accident since there are no design or operating changes involved and the test is not an accident initiator. The proposed extension of the test interval does not involve a significant increase in the consequences since analysis has shown that, the proposed extension of the ILRT and DWBT [drywell bypass test] frequency has a minimal impact on plant risk. 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 extension to the interval for the ILRT does not involve any design or operational changes that could lead to a new or different kind of accident from any accidents previously evaluated. The tests are not being modified, but are only being performed after a longer interval. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Therefore, 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.

An evaluation of extending the ILRT DWBT surveillance frequency from once in 10 years to once in 15 years and 8 months has been performed using methodologies based on the approved ILRT methodologies. This evaluation assumed that the DWBT frequency was being adjusted in conjunction with the ILRT frequency. This analysis used realistic, but still conservative, assumptions with regard to developing the frequency of leakage classes associated with the ILRT and DWBT. The results from this conservative analysis indicates that the proposed extension of the ILRT frequency has a minimal impact on plant risk and therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Thomas G. Hiltz.

Entergy Operations Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: August 2, 2007.

Description of amendment request: The proposed changes to the technical specifications (TSs) will add new analytical methods and modify the containment average air temperature and safety injection tank level to support the implementation of Combustion Engineering 16 x 16 Next Generation Fuel (NGF) as defined in Westinghouse Topical Report WCAP–16500-P beginning in Cycle 16 commencing after the spring 2008 refueling outage. The fuel design is intended to provide improved fuel reliability by reducing grid-to-rod fretting issues, improved fuel performance for high duty operation, and enhanced operating margin.

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.

Core Operating Limits Report (COLR)

The proposed changes to the COLR TS are administrative in nature and have no impact on any plant configuration or system performance relied upon to mitigate the consequences of an accident. Changes to the calculated core operating limits may only be made using NRC approved methodologies, must be consistent with all applicable safety analysis limits, and are controlled by the 10 CFR 50.59 process.

The proposed change will add the following topical reports to the list of referenced core operating analytical methods.

WCAP–16500–P and Final Safety Evaluation (SE)

Westinghouse topical report WCAP–16500–P describes the methods and models that will be used to evaluate the acceptability of CE 16 x 16 NGF at CE plants. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met. Prior to implementation of NGF the new core design will be analyzed with applicable NRC staff approved codes and methods.

WCAP–12610–P–A and CENPD–404–P–A Addendum 1–A

The proposed change allows the use of methods required for the implementation of Optimized ZIRLOTM clad fuel rods. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

WCAP–16523–P and Final Safety Evaluation

This topical report describes the departure from nucleate boiling correlations that will be used to account for the impact of the CE 16 x 16 NGF fuel assembly design. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met. Prior to implementation of NGF the new core design will be analyzed with applicable NRC staff approved codes and methods.

CENPD–387–P–A

The proposed addition of this topical report provides the departure from nucleate boiling (DNB) correlation that will be used to evaluate the DNB impact of non-mixing vane grid spans for CE 16 x 16 standard and NGF assemblies Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

CENPD–132, Supplement 4–P–A, Addendum 1–P and Final Safety Evaluation

The addendum provides an optional steam cooling model that can be used for Emergency Core Cooling System (ECCS) Performance analyses to support the implementation of the CE 16 x 16 NGF fuel assembly design. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

Assumptions used for accident initiators and/or safety analysis acceptance criteria are not altered by the addition of these topical reports.

Safety Injection Tank Water Level and Containment Average Air Temperature

These values are used as inputs to the LBLOCA and SBLOCA analyses. The new limits ensure that the analyzed LBLOCA remain acceptable. The limits have no impact to the SBLOCA analysis results. The changes do not cause an increase in the probability of an accident or an increase in the dose consequences associated with a LBLOCA.

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.

Core Operating Limits Report (COLR)

The proposed change identifies changes in the codes used to confirm the values of selected cycle-specific reactor physics parameter limits. The proposed change allows the use of methods required for the implementation of CE 16 x 16 NGF. The proposed addition of the referenced topical reports has no impact on any plant configurations or on system performance that is relied upon to mitigate the consequences of an accident. The change to the COLR is administrative in nature and does not result in a change to the physical plant or to the modes of operation defined in the facility license.

WCAP–16500–P and Final Safety Evaluation

The proposed change adds Westinghouse topical report WCAP–16500–P, which describes the methods and models that will be used to evaluate the acceptability of CE 16 x 16 NGF at CE plants. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met. Prior to implementation of NGF, the new core design will be analyzed with applicable NRC staff approved codes and methods.

WCAP–12610–P–A and CENPD–404–P–A Addendum 1–A

The proposed change allows the use of methods required for the implementation of Optimized ZIRLOTM clad fuel rods. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

WCAP–16523–P and Final Safety Evaluation

This topical report describes the departure from nucleate boiling correlations that will be used to account for the impact of the CE 16 x 16 NGF fuel assembly design. Entergy has demonstrated that the Limitations and Conditions associated with the SE will be met.

CENPD–387–P–A

The proposed addition of this topical report provides the departure from nucleate boiling (DNB) correlation that will be used to evaluate the DNB impact of non-mixing vane grid spans for CE 16 x 16 standard and NGF assemblies. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

CENPD–132, Supplement 4–P–A, Addendum 1–P and Final Safety Evaluation

The addendum provides an optional steam cooling model that can be used for ECCS Performance analyses to support the implementation of the CE 16 x 16 NGF fuel assembly design. Entergy has demonstrated that the Limitations and Conditions associated with the NRC SE will be met.

Safety Injection Tank Water Level and Containment Average Air Temperature

The safety injection tank (SIT) system provides a passive means of adding a large quantity of borated water to the reactor core in the event of a LBLOCA. The SIT system serves no other purpose. Reducing the maximum volume will not create any new or different accidents.

The containment average air temperature ensures that the peak cladding temperature and cladding oxidation remain within limits during a LBLOCA. The change in the minimum allowable containment average temperature does not create any new or different accidents.

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.

Core Operating Limits Report (COLR)

The addition of the following topical reports to the list of analytical methods referenced in the COLR is administrative in nature:

- WCAP–16500–P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) WCAP–16500–P, Revision 0, ''CE [Combustion Engineering] 16x16 Next Generation Fuel [(NGF)] Core Reference Report''
- WCAP–12610–P–A and CENPD–404–P–A Addendum 1–A
- WCAP–16523–P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR), WCAP–16523–P, ''Westinghouse Correlations WSSV and WSSV–T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes''
- CENPD–387–P–A
- CENPD–132, Supplement 4–P–A, Addendum 1–P and Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) CENPD–132 Supplement 4–P–A, Addendum 1–P, ''Calculative Methods for the CE [Combustion Engineering] Nuclear Power Large Break LOCA Evaluation Model—Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood''

Safety Injection Tank Water Level and Containment Average Air Temperature

The change to the allowable range for these two parameters does not reduce a margin of safety. The changes add to the margin of safety and provide assurance that the peak cladding temperature and cladding oxidation remain within limits.

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

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

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

NRC Branch Chief: Thomas G. Hiltz.

Exelon Generation Company, LLC, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois; Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Units 1 and 2, Will County, Illinois

Date of amendment request: July 31, 2007.

Description of amendment request: The proposed amendment would revise Technical Specification 5.5.2, ''Primary Coolant Sources Outside Containment,'' to clarify the intent of refueling cycle intervals (i.e., 18 month intervals) with respect to system integrated leak test requirements and to add a statement that the provisions of Surveillance Requirement 3.0.2 are applicable.

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 amendment affects only the interval at which integrated system leak tests are performed, not the effectiveness of the integrated system leak test requirements. Revising the integrated system leak test requirements from ''at refueling cycle interval or less'' to ''at least once per 18 months'' is considered to be an administrative change because Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, operate on 18-month fuel cycles. Incorporation of the allowance to extend the 18-month interval by 25%, as allowed by Surveillance Requirement (SR) 3.0.2, does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency.

Test intervals are not considered as initiators of any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased by the proposed amendment. Technical Specification (TS) 5.5.2 continues to require the performance of periodic integrated system leak tests. Therefore, accident analysis assumptions will still be verified. As a result, the consequences of any accident previously evaluated are not significantly increased.

Based on the above discussion, the proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed amendment affects only the interval at which integrated system leak tests are performed; they do not alter the design or physical configuration of the plant. No changes are being made to Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, that would introduce any new accident causal mechanisms.

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

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

Response: No.

The proposed amendment does not change the design or function of plant equipment. The proposed amendment does not significantly reduce the level of assurance that any plant equipment will be available to perform its function.

The proposed amendment provides operating flexibility without significantly affecting plant operation.

Based on this evaluation, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Russell Gibbs.

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

Date of amendment request: June 18, 2007.

Description of amendment request: The proposed amendments would revise Technical Specification 3.7.5, ''Control Room Area Ventilation Air Conditioning (AC) System,'' to add an Action Statement for two inoperable control room area ventilation AC subsystems.

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 proposed change is described in Technical Specification Task Force (TSTF) Standard TS Change Traveler TSTF–477 adds an action statement for two inoperable control room subsystems. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). The proposed changes add an action statement for two inoperable control room subsystems. The equipment qualification temperature of the control room equipment is not affected. Future changes to the Bases or licensee-controlled document will be evaluated pursuant to the requirements of 10 CFR 50.59, ''Changes, Test and Experiments,'' to ensure that such changes do not result in more than a minimal increase in the probability or consequences of an accident previously evaluated. The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not adversely affect the ability of structures, systems and components to perform their intended safety function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological consequences of any accident previously evaluated. Further, the proposed changes do not increase the types and the amounts of radioactive effluent that may be released, nor significantly increase individual or cumulative occupation/public radiation exposures. Therefore, the changes do not involve a significant increase in the probability or 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 proposed changes add an action statement for two inoperable control room subsystems. The changes do not involve a physical altering of the plant (i.e., no new or different type of equipment will be installed) or a change in methods governing normal plant operation. The requirements in the TS continue to require maintaining the control room temperature within the design limits. Therefore, the changes do 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 proposed changes add an action statement for two inoperable control room subsystems. Instituting the proposed changes will continue to maintain the control room temperature within design limits. Changes to the Bases or license controlled document are performed in accordance with 10 CFR 50.59. This approach provides an effective level of regulatory control and ensures that the control room temperature will be maintained within design limits. The proposed changes maintain sufficient controls to preserve the current margins of safety.

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

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

NRC Branch Chief: Russell Gibbs.

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

Date of amendment request: June 27, 2007.

Description of amendment request: The proposed amendment would remove the operability and surveillance requirements for the drywell air temperature and suppression chamber air temperature instrumentation from the Limerick Generating Station (LGS) technical specifications. This will allow a relocation of these requirements to the LGS technical requirements manual, a licensee controlled document.

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 failure of the drywell air temperature or suppression chamber air temperature instrumentation is not assumed to be an initiator of any analyzed event in the UFSAR [Updated Final Safety Analysis Report]. The proposed changes do not alter the physical design of this instrumentation or any other plant structure, system, or component. The proposed changes relocate the drywell air temperature and suppression chamber air temperature instrumentation operability and surveillance requirements from the Limerick Generating Station (LGS) Technical Specifications (TS) to a licensee-controlled document under the control of 10 CFR 50.59 [Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Section 50.59].

The proposed changes conform to NRC regulatory requirements regarding the content of plant TS as identified in 10 CFR 50.36, and also the guidance as approved by the NRC in NUREG–1433, ''Standard Technical Specifications-General Electric Plants, BWR/4.''

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

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

Response: No.

The proposed changes relocate the drywell air temperature and suppression chamber air temperature instrumentation operability and surveillance requirements from the LGS TS to a licensee-controlled document under the control of 10 CFR 50.59. The proposed changes do not alter the physical design, safety limits, or safety analysis assumptions associated with the operation of the plant. Accordingly, the proposed changes do not introduce any new accident initiators, nor do they reduce or adversely affect the capabilities of any plant structure, system, or component in the performance of their safety function.

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

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

Response: No.

The subject instrumentation does not provide primary information required to permit operators to take specific manually controlled actions for which no automatic control is provided, and that are required for safety systems to accomplish their safety functions for design basis accident events. The instrumentation provides only drywell air temperature indication and suppression chamber air temperature indication, and does not provide an input to any automatic safety function. Operability and surveillance requirements will be established in a licensee-controlled document to ensure the reliability of drywell air temperature and suppression chamber air temperature instrumentation capability. Changes to these requirements will be subject to the controls of 10 CFR 50.59, providing the appropriate level of regulatory control.

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

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

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

NRC Branch Chief: Harold K. Chernoff.

FirstEnergy Nuclear Operating Company, et. al., Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit No. 1, Ottawa County, Ohio

Date of amendment request: April 12, 2007.

Description of amendment request: The proposed amendment request would make the operating license and technical specification changes necessary to allow an increase in the rated thermal power from 2772 megawatts thermal (MWt) to 2817 MWt (approximately 1.63 percent), based on the use of Caldon, Inc. Leading Edge Flow Meter CheckPlus™ System instrumentation to improve the accuracy of the plant power calorimetric measurement.

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.

Under contract to the FirstEnergy Nuclear Operating Company, AREVA NP Inc. performed evaluations of the Davis-Besse Nuclear Power Station (DBNPS) Nuclear Steam Supply System (NSSS) and balance of plant systems, components, and analyses that could be affected by the proposed change to the licensed power level. A power uncertainty calculation was performed and the effect of increasing core thermal power by 1.63 percent to 2817 MWt on the DBNPS design and licensing basis was evaluated. The evaluations determined that all structures, systems and components will continue to be capable of performing their design function at the proposed uprated power level of 2817 MWt. An evaluation of the accident analyses demonstrates that the applicable analysis acceptance criteria continue to be met with the proposed changes. No accident initiators are affected by the power uprate and no challenges to any plant safety barriers are created by any of the proposed changes.

The proposed change to the licensed power level does not affect the release paths, the frequency of release, or the analyzed source term for any accidents previously evaluated in the DBNPS Updated Final Safety Analysis Report (UFSAR). Systems, structures, and components required to mitigate transients will continue to be capable of performing their design functions with the proposed changes, and thus were found acceptable. The reduced uncertainty in the power calorimetric measurement ensures that applicable accident analyses acceptance criteria will continue to be met with operation at the proposed power level of 2817 MWt. Analyses performed to assess the effects of mass and energy remain valid. The source term used to assess radiological consequences has been reviewed and determined to bound operation at the proposed power level.

The proposed change to the RPS high flux setpoint Allowable Value does not alter the typical manner in which systems or components are operated, and, therefore, will not result in an increase in the probability of an accident. The proposed High Flux Trip Allowable Values preserve assumptions of current accident analyses at the higher thermal power allowed by the proposed amendment, irrespective of the source of Heat Balance calculation input data. This proposed change does not alter any assumption previously made in the radiological consequence evaluations, nor does it affect mitigation of the radiological consequences of an accident previously evaluated. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The addition of references to Note 10 to Functional Unit 2, High Flux, in Table 4.3–1 is administrative and does not impact the probability or consequences of an accident previously evaluated because its inclusion does not involve an accident initiator or impact any radiological analyses. This change is made to incorporate NRC guidance in a manner previously determined to be acceptable in DBNPS License Amendment No. 274.

The proposed change to the volume of the condensate storage tanks does not alter the typical manner in which the system or component is operated, and, therefore, will not result in a significant increase in the probability of an accident. The condensate storage tanks are not accident initiators. The proposed change preserves the assumptions previously made in the radiological consequence evaluations and the radiological consequences of accidents previously evaluated. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to the Core Operating Limits Report (COLR) portion of the Administrative Controls Section of the TS are administrative and do not impact the probability or consequences of an accident previously evaluated because their inclusion do not involve accident initiators or impact any radiological analyses. These changes are made to include the NRC-approved documents pertaining to the Caldon Leading Edge Flow Meter.

In summary, none of the proposed changes 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.

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of any of the proposed changes. Use of the Caldon CheckPlus™ System has been analyzed, and failures of the system will have no adverse effect on any safety-related system or any systems, structures, and components required for transient mitigation. Systems, structures, and components previously required for the mitigation of a transient continue to be capable of fulfilling their intended design functions. The proposed changes have no significant adverse affect on any safety-related structures, systems or components and do not significantly change the performance or integrity of any safety-related system.

The proposed changes do not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than previously evaluated. Operating at a core power level of 2817 MWt does not create any new accident initiators or precursors. The reduced uncertainty in the power calorimetric measurement ensures that applicable accident analyses acceptance criteria continue to be met, to support operation at the proposed core power level of 2817 MWt. Credible malfunctions continue to be bounded by the current accident analyses of record or recent evaluations that demonstrate that applicable criteria will continue to be met with the proposed changes.

The proposed change to the RPS high flux setpoint Allowable Value does not introduce new accident scenarios, failure mechanisms or single failures. The change does not alter the manner in which plant systems or components are operated. The proposed High Flux Trip Allowable Values preserve assumptions of current accident analyses at the higher thermal power allowed by the proposed amendment, irrespective of the source of Heat Balance calculation input data. Therefore, this proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The addition of a reference to Note 10 to Functional Unit 2, High Flux, in Table 4.3–1 is administrative and will not create the possibility of a new or different kind of accident from any accident previously evaluated because its inclusion will not change the manner in which any equipment is operated. The proposed change to the volume of the condensate storage tanks does not introduce new accident scenarios, failure mechanisms or single failures. Therefore, this proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to the COLR portion of the Administrative Controls Section of the TS are administrative and will not create the possibility of a new or different kind of accident from any accident previously evaluated because their inclusion will not change the manner in which any equipment is operated.

In summary, none of the proposed changes will 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 margins of safety associated with the power uprate are those pertaining to core thermal power. These include those associated with the fuel cladding, Reactor Coolant System pressure boundary, and containment barriers. An engineering evaluation of the proposed 1.63 percent increase in core thermal power was performed. The power uprate required revised NSSS design thermal and hydraulic parameters to be established to serve as the basis for all of the NSSS analyses and evaluations. This engineering review identified the design modifications necessary to accommodate the revised NSSS design conditions. Evaluations determined that the NSSS systems and components will continue to operate satisfactorily at the uprated power level with these modifications and the proposed changes. The NSSS accident analyses were evaluated at the uprated power level. In all cases, the evaluations demonstrate that the applicable analyses acceptance criteria will continue to be met with approval of the proposed changes. As such, the margins of safety will continue to be bounded by the analyses for all the changes being proposed.

Therefore, none of the proposed changes will involve a significant reduction in a margin of safety.

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

Attorney for licensee: David W. Jenkins, Attorney, FirstEnergy Corporation, Mail Stop A–GO–15, 76 South Main Street, Akron, OH 44308.

NRC Branch Chief: Russell Gibbs.

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

Date of amendment request: April 25, 2007, as supplemented by letter dated June 28, 2007.

Description of amendment request: The proposed amendment would change the operating license and technical specifications to increase the maximum power level from 2568 megawatts thermal (MWt) to 2609 MWt. The approximately 1.6 percent increase in power level would be achieved by use of the Caldon Leading Edge Flowmeter CheckPlus system to accurately measure power level.

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

The proposed change will increase the maximum core power level from 2568 MWt to 2609 MWt. This increase will only require adjustments and calibrations of existing plant instrumentation and control systems. The only equipment upgrades necessary for this uprate are spool pieces containing multiple ultrasonic flow instruments, which will be installed in each feedwater line, as well as more accurate instrumentation for feedwater pressure and steam pressure and temperature. Indication and control functions will continue to be performed by the currently installed feedwater instrumentation.

Nuclear steam supply systems (NSSS) and balance-of-plant (BOP) systems and components that could be affected by the proposed change have been evaluated using revised NSSS design parameters based on a core power level of 2609 MWt. The results of these evaluations, which used well-defined analysis input assumptions/parameter values and currently approved analytical techniques, indicate that CR–3 systems and components will continue to function within their design parameters and remain capable of performing their required safety functions at 2609 MWt. Since the revised NSSS parameters remain within the design conditions of the Reactor Coolant System (RCS) functional specification, the proposed change will not result in any new design transients or adversely affect the current CR–3 design transient analyses.

The accidents analyzed in Chapter 14 of the CR–3 Final Safety Analysis Report (FSAR) have been reviewed for the impact of the uprate. Based on the power levels assumed in the current safety analyses, it has been determined that all FSAR and supporting analyses bound the uprate. This includes the dose calculations for the design basis radiological accidents, which assume a power level of 2619 MWt (2568 MWt plus an assumed 2 percent measurement uncertainty). Since the proposed change relies on less than 0.4% uncertainty, the assumed power level of 100.4% of 2609 MWt remains 2619 MWt. Therefore, analyses performed at this power remain bounding.

(2) Does not create the possibility of a new or different kind of accident from any accident previously evaluated.

As discussed above, the only equipment upgrades necessary for this uprate are spool pieces containing multiple ultrasonic flow instruments, which will be installed in each feedwater line, as well as more accurate instrumentation for feedwater pressure and steam pressure and temperature. All CR–3 systems and components will continue to function within their design parameters and remain capable of performing their required safety functions. The proposed change does not impact current CR–3 design transients or introduce any new transients. Equipment failure modes are expected to be the same as for existing instruments. Protective and control functions will continue to be performed by the currently installed feedwater instrumentation. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does not involve a significant reduction in a margin of safety

Challenges to the fuel, RCS pressure boundary and containment were evaluated for uprate conditions. Core analyses show that the implementation of the power uprate will continue to meet the current nuclear design basis. Impacts to components associated with RCS pressure boundary structural integrity, and factors such as pressure/temperature limits, vessel fluence, and pressurized thermal shock (PTS) were determined to be bounded by current analyses.

As discussed above, all systems will continue to operate within their design parameters and remain capable of performing their intended safety functions following implementation of the proposed change. Finally, the current CR–3 safety analyses, including the design basis radiological accident dose calculations, bound the uprate.

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

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

Attorney for licensee: David T. Conley, Associate General Counsel II—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Branch Chief: Thomas H. Boyce.

Nine Mile Point Nuclear Station (NMPNS), LLC, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1 (NMP1), Oswego County, New York

Date of amendment request: July 12, 2007.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) requirements related to control room envelope (CRE) habitability in TS 3.4.5, ''Control Room Air Treatment System,'' and TS 6.5, ''Programs and Manuals.'' The proposed changes are consistent with TS Task Force (TSTF) change 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 line item improvement process. The licensee affirmed the applicability of the model no significant hazards consideration determination in its application.

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 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 [a] 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 proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

NRC Branch Chief: Mark G. Kowal.

Nine Mile Point Nuclear Station (NMPNS), LLC, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1 (NMP1), Oswego County, New York

Date of amendment request: July 23, 2007.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) section 3.1.1, ''Control Rod System,'' to incorporate a provision that should the rod worth minimizer (RWM) become inoperable before a reactor startup is commenced or before the first 12 control rods have been withdrawn, startup would be allowed to continue. This provision would rely on the RWM function being performed manually and would require a double check of compliance with the control rod program by a second licensed operator or other qualified member of the technical staff. The use of this allowance would be limited to one startup in the last calendar year.

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 allows plant startup to proceed if the RWM becomes inoperable prior to withdrawing the first 12 control rods. The relevant design basis accident is the control rod drop accident (CRDA), which involves multiple failures to initiate the event. This change does not increase the probability of occurrence of any of the failures that are necessary for a CRDA to occur. Use of the RWM or the alternate use of a second qualified individual to ensure the correct control rod withdrawal sequence is not in itself an accident initiator, and adding the new startup allowance does not involve any plant hardware changes or new operator actions that could serve to initiate a CRDA. The proposed change will have no adverse effect on plant operation, or the availability or operation of any accident mitigation equipment. Also, since the control rod program will continue to be enforced by either the RWM or verification by a second qualified individual, the initial conditions of the CRDA radiological consequence analysis presented in the Updated Final Safety Analysis Report are not affected. Therefore, there will be no 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 introduce any new modes of plant operation and will not result in a change to the design function or operation of any structure, system, or component that is used for accident mitigation. The proposed change allows plant startup to proceed if the RWM becomes inoperable prior to withdrawing the first 12 control rods, with verification of control rod movement in the correct sequence performed by a second qualified individual. This change does not result in any credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing basis. This change does not affect the ability of safety-related systems and components to perform their intended safety functions. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any [accident] previously evaluated.

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

Response: No.

The proposed change allows plant startup to proceed if the RWM becomes inoperable prior to withdrawing the first 12 control rods. The proposed change will have no adverse effect on plant operation or equipment important to safety. The relevant design basis accident is the [CRDA], which involves multiple failures to initiate the event. The CRDA analysis consequences and related initial conditions remain unchanged when invoking the proposed change. The plant response to the CRDA will not be affected and the accident mitigation equipment will continue to function as assumed in the accident analysis. Therefore, there will be no significant reduction in a 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 request involves no significant hazards consideration.

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

NRC Branch Chief: Mark G. Kowal.

Nine Mile Point Nuclear Station (NMPNS), LLC, Docket No. 50–410, Nine Mile Point Nuclear Station Unit No. 2 (NMP2), Oswego County, New York

Date of amendment request: July 12, 2007.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) requirements related to control room envelope (CRE) habitability in TS 3.7.2, ''Control Room Envelope Filtration (CREF) System,'' and TS 5.5, ''Programs and Manuals.'' The proposed changes are consistent with TS Task Force (TSTF) change 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 line item improvement process. The licensee affirmed the applicability of the model no significant hazards consideration determination in its application.

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 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 [a] 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 proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

NRC Branch Chief: Mark G. Kowal.

Nine Mile Point Nuclear Station (NMPNS), LLC, Docket No. 50–410, Nine Mile Point Nuclear Station Unit No. 2 (NMP2), Oswego County, New York

Date of amendment request: July 30, 2007.

Description of amendment request: The proposed amendment would revise the technical specifications (TSs) by changing the testing frequency for drywell spray nozzles specified in TS Surveillance Requirement (SR) 3.6.1.6.3 from ''10 years'' to ''following maintenance that could result in nozzle blockage.''

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 modifies the surveillance requirement (SR) to verify that the drywell spray nozzles are unobstructed after maintenance that could introduce material that could result in nozzle blockage. The spray nozzles are not assumed to be initiators of any previously analyzed accident. Therefore, the proposed change does not increase the probability of any accident previously evaluated. The spray nozzles are used in the accident analyses to mitigate design basis accidents. The revised SR to verify system operability following maintenance is considered adequate to ensure operability of the Residual Heat Removal (RHR) Drywell Spray System.

Since the system will still be able to perform its accident mitigation function, the consequences of accidents previously evaluated are not increased.

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

2. 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 revises the SR to verify that the RHR Drywell Spray System nozzles are unobstructed after maintenance that could result in nozzle blockage. The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators or impact the assumptions made in the safety analysis.

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

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

Response: No.

The proposed change revises the frequency for performance of the SR to verify that the RHR Drywell Spray System nozzles are unobstructed. The frequency is changed from every 10 years to following maintenance that could result in nozzle blockage. This requirement, along with the foreign material exclusion program, the normal environmental conditions for the system, and the remote physical location of the spray nozzles, provide assurance that the spray nozzles will remain unobstructed. As the spray nozzles are expected to remain unobstructed and able to perform their post-accident mitigation function, plant safety is not significantly affected.

Therefore, the proposed change does not involve a significant reduction in a 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 request involves no significant hazards consideration.

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

NRC Branch Chief: Mark G. Kowal.

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

Date of amendment request: June 29, 2007.

Description of amendment request: The proposed amendments would modify the Technical Specifications (TSs) 3.7.2, by removing the specific isolation time for the main steam isolation valves from the associated TS Surveillance Requirements (SRs) and by replacing it with the requirement to verify the valve isolation time is within limits. The changes are consistent with Nuclear Regulatory Commission (NRC) approved Industry/Technical Specification Task Force (TSTF)–491, "Removal of the Main Steam and Main Feedwater Valve Isolation Time from Technical Specifications," Revision 2. The proposed amendments deviate from TSTF–491 in that the current PBNP TS 3.7.3, and associated SRs do not include the main feedwater valve closure times, and thus TSTF–491 changes to TS 3.7.3 are not applicable to the PBNP TSs.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on October 5, 2006 (71 FR 58884), on possible amendments concerning the Consolidated Line Item Improvement Process (CLIIP), including a model safety evaluation and a model no significant hazards consideration determination. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on December 29, 2006 (71 FR 78472) as part of the CLIIP. In its application dated June 29, 2007, the licensee affirmed the applicability of the following determination.

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 proposed change allows relocating main steam [ ] valve isolation times to the Licensee Controlled Document that is referenced in the Bases. The proposed change is described in Technical Specification Task Force (TSTF) Standard TS Change Traveler TSTF–491 related to relocating the main steam [ ] valve isolation times to the Licensee Controlled Document that is referenced in the Bases and replacing the isolation time with the ph[r]ase, "within limits."

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). The proposed changes relocate the main steam [ ] isolation valve times to the Licensee Controlled Document that is referenced in the Bases. The requirements to perform the testing of these isolation valves are retained in the TS. Future changes to the Bases or licensee-controlled document will be evaluated pursuant to the requirements of 10 CFR 50.59, "Changes, test and experiments," to ensure that such changes do not result in more than minimal increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not adversely affect the ability of structures, systems and components (SSCs) to perform their intended safety function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological consequences of any accident previously evaluated. Further, the proposed changes do not increase the types and the amounts of radioactive effluent that may be released, nor significantly increase individual or cumulative occupation/public radiation exposures.

Therefore, the changes do not involve a significant increase in the probability or 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 proposed changes relocate the main steam [ ] valve isolation times to the Licensee Controlled Document that is referenced in the Bases. In addition, the valve isolation times are replaced in the TS with the ph[r]ase "within limits." The changes do not involve a physical altering of the plant (i.e., no new or different type of equipment will be installed) or a change in methods governing normal p[l]ant operation. The requirements in the TS continue to require testing of the main steam [ ] isolation valves to ensure the proper functioning of these isolation valves.

Therefore, the changes do 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 proposed changes relocate the main steam [ ] valve isolation times to the Licensee Controlled Document that is referenced in the Bases. In addition, the valve isolation times are replaced in the TS with the ph[r]ase "within limits." Instituting the proposed changes will continue to ensure the testing of main steam [ ] isolation valves. Changes to the Bases or license controlled document are performed in accordance with 10 CFR 50.59. This approach provides an effective level of regulatory control and ensures that main steam [ ] isolation valve testing is conducted such that there is no significant reduction in the margin of safety.

The margin of safety provided by the isolation valves is unaffected by the proposed changes since there continue to be TS requirements to ensure the testing of main steam [ ] isolation valves. The proposed changes maintain sufficient controls to preserve the current margins of safety.

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

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

NRC Acting Branch Chief: Travis L. Tate.

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

Date of amendment request: August 15, 2007.

Description of amendment request: The proposed amendment would revise the licensing basis, as described in Appendix 3A of the Salem Updated Final Safety Analysis Report (UFSAR), regarding the method of calculating the net positive suction head available (NPSHa) for the emergency core cooling system (ECCS) and containment heat removal system pumps. The proposed change relates to issues associated with Generic Letter 2004–02, ''Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors.''

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 change in NPSH methodology for ECCS pumps allows the use of initial containment air pressure in calculating NPSHa. Although this change is a non-conservative change in the Salem methodology for calculation of RHR [residual heat removal] pump NPSHa during post LOCA [loss-of-coolant accident] recirculation (per 10 CFR 50.59(c)(1)(viii) [Title 10 of the Code of Federal Regulations, Part 50, Section 50.59(c)(1)(viii)]), the proposed new methodology is in accordance with NPSHa calculation methodologies provided in Safety Guide 1, Regulatory Guides [RG] 1.1, and 1.82, and the guidance of NEI [Nuclear Energy Institute] 04–07, [''Pressurized Water Reactor Sump Performance Evaluation Methodology[,''] (GSI [generic safety issue]—191) and accompanying SER [safety evaluation report]. The containment air pressure value used in the NPSHa calculation is based on the containment conditions prior to the accident only and does not include any credit for accident pressure conditions, is conservatively determined based on minimum containment initial pressure, and maximum temperature and relative humidity conditions. In addition, the vapor pressure term for the sump water being pumped is also included in the NPSHa equation, and the value chosen for the NPSHa calculation is based on the highest temperature of the sump fluid for the condition being evaluated. This, in conjunction with the more rigorous GSI–191 analyses, provides assurance that the ECCS pumps can perform their design function. Consequently, the ECCS pumps will continue to perform their design function and there is no 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 ECCS pumps take suction from the containment sump during the recirculation phase of the LOCA to provide long term core cooling. This system is not utilized during normal operation of the plant. Therefore, it does not cause initiation of any accident.

However, the ECCS pumps will continue to perform their design function during the recirculation phase. Crediting initial containment air pressure in the NPSH methodology does not create any new or different kind of accident from any accident previously evaluated. This change removes an additional conservatism built into the original methodology. By changing the UFSAR described methodology to credit the containment initial air pressure in the RHR pump NPSHa calculation, a more realistic methodology is established. The sole purpose of the additional conservatism was to ensure credit was not taken for post-LOCA pressure. The revised methodology continues to meet this requirement.

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

Response: No.

The proposed change removes conservatism from the existing UFSAR methodology. However, the purpose of the conservatism (equating containment pressure to sump vapor pressure) was solely to ensure that no credit was taken for transient (post-LOCA) pressure in the NPSHa calculation. The purpose was not to deny credit for initial containment air pressure. Consequently, removing the conservatism does not alter the basic intent of the NPSH methodology per RG 1.1 requirements, and is consistent with the requirements of RG 1.82, Revision 1 and NEI 04–07. This change to include a containment air pressure value establishes a more realistic methodology that still encompasses adequate conservatisms; no credit is given for the higher accident pressure conditions, and the value is conservatively determined based on minimum initial containment air pressure and maximum temperature and relative humidity conditions. In addition, the vapor pressure term for the sump water being pumped is also added to the NPSHa equation, and the value chosen for the NPSHa calculation is based on the highest temperature of the sump fluid for the condition being evaluated. Consequently, this change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Harold K. Chernoff.

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

Date of amendment request: August 20, 2007.

Description of amendment request: The amendment would increase the minimum volume of fuel required for the emergency diesel generators (EDGs) in Technical Specification (TS) 3.8.3, ''Diesel Fuel Oil, Lube Oil, and Starting Air.''

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 changes to the minimum required fuel oil volume required in the EDG storage tanks have no impact on the frequency of occurrence of any of the accidents evaluated in the FSAR [Final Safety Analysis Report for Callaway]. Changing the minimum required fuel oil volume in the EDG fuel oil storage tank has no impact on the likelihood of occurrence of a loss of coolant accident (LOCA), line break, plant transient, loss of offsite power, or any such accident because the precursors for such accidents do not involve the fuel oil storage tanks.

The EDGs are designed to provide [alternating current] electrical power to systems required for mitigating the effects of accidents in the event of a loss of the preferred (offsite) power source (i.e., from the grid). However, the failure or malfunction of an EDG (due, for example, to a loss or interruption of [the] fuel oil supply) is not itself an initiator of any accident previously evaluated.

Based on these considerations, the proposed changes have no impact on the probability of occurrence of any accident evaluated in the FSAR, and therefore the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

With respect to the consequences of postulated accidents addressed in [the] FSAR, the support function provided by the EDGs for accident mitigation is not affected by the proposed TS changes. [The proposed changes are to provide additional margin for precluding adverse effects that could result from air entrapment caused by a vortex condition during fuel oil transfer pump operation and, thus, to ensure that the EDG has sufficient fuel oil to provide its support function when needed.] Each of the diesel fuel oil storage tanks has adequate excess capacity to more than accommodate a slight increase in the usable volume of fuel oil contained therein. Thus, even with this increase, the tanks will still be fully capable of storing the required fuel oil volume needed to ensure EDG operation throughout the assumed duration of an accident. At the same time, the proposed changes to TS 3.8.3 will serve to ensure that the unusable volume in the tanks provides adequate margin against potentially adverse vortex effects (by precluding the potential for air ingestion into the fuel oil transfer pumps). On this basis, the proposed changes have no impact on the capability of the EDGs to perform their required mitigation/support function for accidents involving a loss of offsite power. Since the proposed changes have no impact on accident mitigation capability, they involve no increase in the consequences of any accident evaluated in the FSAR.

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

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

Response: No.

The proposed changes involve a slight change to the minimum fuel oil volume required for the EDGs, but they do not involve hardware changes or changes to EDG operation or testing that would create any new failure modes for the EDGs or any other [safety-related] system or component, or that would adversely affect plant operation. The changes do not involve the addition of any new equipment. No changes to accident assumptions, including any new limiting single failures, are involved. With respect to the proposed changes, the plant will continue to be operated within the envelope of the existing safety analyses.

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

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

Response: No.

The margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The proposed changes do not directly affect these barriers, nor do they involve or cause any adverse impact on the EDGs which serve to support these barriers in the event of an accident concurrent with a loss of offsite power.

[The margin of safety is also related to the ability of the safety-related systems to perform their safety function as described in the safety analyses in the FSAR. The proposed changes are to provide additional margin for precluding adverse effects that could result from air entrapment caused by a vortex condition during fuel oil transfer pump operation and, thus, to ensure that the EDG has sufficient fuel oil to provide its support function when needed. Therefore, the proposed changes are to increase margin for the EDGs.]

The proposed changes do not alter the manner in which safety limits or limiting safety system settings are determined, nor is [the] basis of any limiting condition for operation changed or affected. The safety analysis acceptance criteria are not impacted by these changes. The proposed changes will not result in plant operation in a configuration outside the design basis.

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

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

Attorney for licensee: John O'Neill, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, NW., Washington, DC 20037.

NRC Branch Chief: Thomas G. Hiltz.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: March 1, 2007.

Brief description of amendment: The amendment revised the Grand Gulf Nuclear Station, Unit 1 (GGNS) Technical Specification (TS) to add a note to the Required Actions of TS 3.6.1.3, ''Primary Containment Isolation Valves (PCIVs)''. GGNS TS 3.6.1.3 requires specific actions to be taken for inoperable PCIVs. The TS Required Actions include isolating the affected penetration by use of a closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. The new note would allow a relief valve to be used to comply with

TS 3.6.1.3, Actions A.1 and B.1 without being deactivated provided it has a relief setpoint of at least 1.5 times containment design pressure (i.e., at least 23 pounds per square inch gauge) and meets one of the following criteria:

1. The relief valve is one-inch nominal size or less, or

2. The flow path is into a closed system whose piping pressure rating exceeds the containment design pressure rating.

Date of issuance: August 24, 2007.

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

Amendment No: 176.

Facility Operating License No. NPF–29: The amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in Federal Register: April 24, 2007 (72 FR 20382).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 24, 2007.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket Nos. 50–247 and 50–286, Indian Point Nuclear Generating Unit Nos. 2 and 3, Westchester County, New York

Date of application for amendment: September 25, 2006, as supplemented March 12, 2007.

Brief description of amendment: Entergy Nuclear Operations, Inc., requested an amendment to make editorial changes to the Technical Specifications of Indian Point Nuclear Generating Unit Nos. 2 and 3. The editorial changes consist of typographical corrections, update of references, and deletion of obsolete notes.

Date of issuance: August 16, 2007.

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

Amendment No.: 252 and 234.

Facility Operating License Nos. DPR–26 and DPR–64: The amendment revised the License and the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 7, 2006 (71 FR 65142).

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

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

No significant hazards consideration comments received: No.

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

Date of amendment request: August 7, 2006, as supplemented by letters dated January 22, May 14, and August 7, 2007.

Description of amendment request: The amendment revises the Seabrook Technical Specifications (TSs) to correct a joint-owner name in the operating license, remove a license condition from Appendix C to the FOL, and remove the list of Bases sections from the TS Index. Additionally, the amendment removes two manual valves from TS table 3.3–9, ''Remote Shutdown System,'' adds the requirement that only one charging pump is permitted to be aligned for injection into the reactor coolant system in Modes 4, 5, and 6, removes a 1-hour reporting requirement for portable makeup pump system storage from TS 3.7.4, ''Service Water System/Ultimate Heat Sink,'' deletes a footnote from TS 3.7.6.2, ''Air Conditioning,'' and modifies TS 6.7.6, ''Radioactive Effluent Controls Program.''

Date of issuance: August 23, 2007.

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

Amendment No.: 116.

Facility Operating License No. NPF 86: The amendment revised the License and Technical Specification.

Date of initial notice in ***Federal Register:*** June 5, 2007 (72 FR 31101).

The licensee's January 22, May 14, and August 7, 2007, supplements provided clarifying information that did not change the scope of the proposed amendment as described in the original notice of proposed action published in the **Federal Register**, and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 23, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 14, 2006, as supplemented by letter dated July 16, 2007.

Brief description of amendments: The amendments make miscellaneous improvements to the Technical Specifications (TSs) for Prairie Island Nuclear Generating Plant, Units 1 and 2. The amendments revise the wording in the section headers in TS 1.3, ''Completion Times''; remove an unnecessary Note in TS 3.1.4, ''Rod Group Alignment Limits''; remove applicable modes in TS 3.3.7, ''Spent Fuel Pool Special Ventilation System (SFPSVS) Actuation Instrumentation''; add reference to a TS Condition to clarify the requirements of TS 3.7.10, ''Control Room Special Ventilation System (CRSVS)''; and update a reference in TS 4.0, ''Design Features.''

Date of issuance: August 10, 2007.

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

Amendment Nos.: 180 & 170.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revised the TSs.

Date of initial notice in ***Federal Register:*** November 21, 2006 (71 FR 67397).

The supplemental letter contained clarifying information and did not change the initial no significant hazards consideration determination, and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 10, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 26, 2006, as supplemented on May 14, 2007.

Brief description of amendments: The amendments revise Technical Specification (TS) 6.9.1.9, ''Core Operating Limits Report (COLR),'' to remove the revision numbers and dates from the list of topical reports that contain the analytical methods used in the COLR. The Salem Unit 2 amendment also adds a new topical report to the list of COLR methods referenced in TS 6.9.1.9.

Date of issuance: August 23, 2007.

Effective date: The license amendments are effective as of the date of issuance. The Salem Unit 1 amendment shall be implemented prior to restart from the 19th refueling outage in fall 2008. The Salem Unit 2 amendment shall be implemented prior to restart from the 16th refueling outage in spring 2008.

Amendment Nos.: 284 and 267.

Facility Operating License Nos. DPR 70 and DPR–75: The amendments revised the TSs and the License.

*Date of initial notice in **Federal Register:*** November 7, 2006 (71 FR 65143).

The supplement dated May 14, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the **Federal Register** on November 7, 2006 (71 FR 65143).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 23, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 15, 2006, as supplemented by letters dated June 21, and August 23, 2007.

Brief description of amendment: The amendment deletes Technical Specification (TS) Table 3.6.3–1, ''Primary Containment Isolation Valves,'' and relocates the information to the Hope Creek Generating Station Technical Requirements Manual (TRM). The amendment also revises other TS sections that reference TS Table 3.6.3–1.

Date of issuance: August 27, 2007.

Effective date: As of the date of issuance, to be implemented within 90 days. Implementation shall include the relocation of information from the TSs to the TRM as described in the licensee's application dated November 15, 2006, and letters dated June 21, and August 23, 2007.

Amendment No.: 171.

Facility Operating License No. NPF–57: The amendment revised the TSs and the License.

*Date of initial notice in **Federal Register:*** February 13, 2007 (72 FR 6789).

The supplements dated June 21, and August 23, 2007, provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 27, 2007.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 5th day of September, 2007.

For the Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. E7–17864 Filed 9–10–07; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

[Release No. IC–27965; File No. 812–13359]

Financial Investors Variable Insurance Trust et al., Notice of Application September 4, 2007

AGENCY: Securities and Exchange Commission (''SEC'' or the ''Commission'').

ACTION: Notice of application (''Application'') for exemption, pursuant to section 6(c) of the Investment Company Act of 1940, as amended (the ''1940 Act''), from the provisions of sections 9(a), 13(a), 15(a) and 15(b) of the Act and Rules 6e–2(b)(15) and 6e–3(T)(b)(15) thereunder.

Applicants: Ibbotson Conservative ETF Asset Allocation Portfolio, Ibbotson Income and Growth ETF Asset Allocation Portfolio, Ibbotson Balanced ETF Asset Allocation Portfolio, Ibbotson Growth ETF Asset Allocation Portfolio, Ibbotson Aggressive Growth ETF Asset Allocation Portfolio (collectively, the ''Existing Funds''), each a series of Financial Investors Variable Insurance Trust (the ''Trust''), any other series established from time to time under the Trust (collectively with the Existing Funds, the ''Insurance Funds''), and any future investment company that is designed to fund insurance products and for which ALPS Advisers, Inc. (the ''Investment Adviser''), any successor in interest (collectively with the Investment Adviser, the ''Investment Advisers''), or any affiliates of the Investment Advisers may serve as investment manager, investment adviser, subadviser, administrator, principal underwriter or sponsor (funds advised by such Investment Advisers herein also referred to collectively as the ''Insurance Funds'') (the Trust, the Existing Funds, the Insurance Funds, the Investment Adviser, and the Investment Advisers, referred to collectively as the ''Applicants'').

Summary of Application: The Applicants request an order exempting certain life insurance companies on behalf of their separate accounts that currently invest or may hereafter invest in the Insurance Funds to the extent necessary to permit shares of the Existing Funds (the ''Shares'') and the Insurance Funds to be sold to and held by: (i) Separate accounts funding variable annuity contracts and variable life insurance policies (collectively ''Variable Contracts'') issued by both affiliated life insurance companies and unaffiliated life insurance companies; (ii) trustees of qualified group pension and group retirement plans outside of the separate account context (''Qualified Plans''); (iii) separate accounts that are not registered as investment companies under the 1940 Act pursuant to exemptions from registration under section 3(c) of the 1940 Act; (iv) any Adviser to an Insurance Fund that is permitted to hold shares in an Insurance Fund consistent with the requirements of regulations issued by the Treasury Department (individually a ''Treasury Regulation'' and collectively the ''Treasury Regulations''), specifically Treasury Regulation Section 1.817–5 for the purpose of providing seed capital to an Insurance Fund; and (v) any other Participating Insurance Company permitted to hold shares of an Insurance Fund (''General Accounts'').

Filing Date: The Application was filed on January 26, 2007, and amended and restated on May 21, 2007.

Hearing or Notification of Hearing: An order granting the application will be issued unless the Commission orders a hearing. Interested persons may request a hearing by writing to the Secretary of the Commission and serving Applicants with a copy of the request, personally or by mail. Hearing requests should be received by the Commission by 5:30 p.m. on September 26, 2007, and should be accompanied by proof of service on Applicants in the form of an affidavit or, for lawyers, a certificate of service. Hearing requests should state the nature of the requester's interest, the reason for the request, and the issues contested. Persons who wish to be notified of a hearing may request notification by writing to the Secretary of the Commission.

ADDRESSES: The Commission: Secretary, Securities and Exchange Commission, 100 F Street, NE., Washington, DC 20549–1090; Applicants: c/o Jeffrey T. Pike, Esq., Secretary, Financial Investors Variable Insurance Trust, 1290 Broadway, Suite 1100, Denver, Colorado 80203.

FOR FURTHER INFORMATION CONTACT: Jeffrey A. Foor, Senior Counsel, or Zandra Y. Bailes, Branch Chief, Office of Insurance Products, Division of Investment Management, at (202) 551–6795.