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 6 through August 19, 2004. The last biweekly notice was published on August 19, 2004 (69 FR 51487).

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Date of amendment request: June 22, 2004.

Description of amendment request: The proposed amendment would revise Technical Specification 3.1.8, ''Scram Discharge Volume (SDV) Vent and Drain Valves,'' to allow a vent or drain line with one inoperable valve to be isolated instead of requiring the valve to be restored to Operable status within 7 days.

The U.S. Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on February 24, 2003 (68 FR 8637), on possible amendments to revise the action for one or more SDV vent or drain lines with an inoperable valve, 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 April 15, 2003 (68 FR 18294). The licensee affirmed the applicability of the model NSHC determination in its application dated June 22, 2004.

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

A change is proposed to allow the affected SDV vent and drain line to be isolated when there are one or more SDV vent or drain lines with one valve inoperable instead of requiring the valve to be restored to operable status within 7 days. With one SDV vent or drain valve inoperable in one or more lines, the isolation function would be maintained since the redundant valve in the affected line would perform its safety function of isolating the SDV. Following the completion of the required action, the isolation function is fulfilled since the associated line is isolated. The ability to vent and drain the SDV is maintained and controlled through administrative controls. This requirement assures the reactor protection system is not adversely affected by the inoperable valves. With the safety functions of the valves being maintained, the probability or consequences of an accident previously evaluated are not significantly increased.

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

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

The proposed change ensures that the safety functions of the SDV vent and drain valves are fulfilled. The isolation function is maintained by redundant valves and by the required action to isolate the affected line. The ability to vent and drain the SDV is maintained through administrative controls. In addition, the reactor protection system will prevent filling of the SDV to the point that it has insufficient volume to accept a full scram. Maintaining the safety functions related to isolation of the SDV and insertion of control rods ensures that the proposed change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Mr. Thomas S. O'Neill, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60666.

NRC Section Chief: Anthony J. Mendiola.

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1 (TMI–1), Dauphin County, Pennsylvania

Date of amendment request: April 23, 2004.

Description of amendment request: The proposed amendment would delete Technical Specification (TS) Section 6.16, ''Post-Accident Sampling Programs NUREG 0737 (II.B.3, II–F.1.2),'' and the related requirements to maintain a Post-Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, Revision 3, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Access Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the NRC's lessons learned from the accident that occurred at TMI Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TSs for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on March 3, 2003 (68 FR 10052) on possible amendments to eliminate PASS, 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 a license amendment application in the **Federal Register** on May 13, 2003 (68 FR 25664). The licensee affirmed the applicability of the following NSHC determination in its application dated April 23, 2004.

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 PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the TMI–2 accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections. The system was not intended to and does not serve a function for preventing accidents and its elimination would not affect the probability of accidents previously evaluated.

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a PASS provides little actual benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

The regulatory requirements for the PASS can be eliminated without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post-TMI–2 accident guidance through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

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

The elimination of PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radioisotopes within the containment building.

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

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

The elimination of the PASS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety. Methodologies that are not reliant on PASS are designed to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI–2 accident can be adequately met without reliance on a PASS.

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

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

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

Attorney for licensee: Thomas S. O'Neill, Associate General Counsel, AmerGen Energy Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Section Chief: Richard J. Laufer.

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

Date of amendments request: July 26, 2004.

Description of amendments request: The proposed amendments would delete requirements from the Technical Specifications (TS) to maintain hydrogen recombiners and hydrogen and oxygen monitors. Licensees were generally required to implement upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide (RG) 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to combustible gas control were imposed by Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. The revised 10 CFR 50.44, ''Combustible gas control for nuclear power reactors,'' eliminated the requirements for hydrogen recombiners and relaxed safety classifications and licensee commitments to certain design and qualification criteria for hydrogen and oxygen monitors.

The NRC staff issued a notice of availability of a model no significant hazards consideration determination for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model no significant hazards consideration determination in its application dated July 26, 2004.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

NRC Section Chief (Acting): Michael L. Marshall.

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

Date of amendment request: June 21, 2004.

Description of amendment request: The proposed amendment would revise Technical Specification Section 5.5.14, ''Technical Specifications (TS) Bases Control Program,'' to replace the previous 10 CFR 50.59 term ''unreviewed safety question'' with current terminology. The proposed amendment would also revise TS Section 5.7.1, ''High Radiation Area,'' to add wording that was inadvertently deleted with the issuance of the Improved Standard Technical Specifications in Amendment No. 176.

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

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

The proposed changes do not modify the facility or the procedures for operation of the facility. One change updates the terminology used in 10 CFR 50.59 evaluations. The change does not alter the requirement of the TS Bases Control Program. The requirement for NRC review and approval of a TS Bases change is still determined through the use of the 10 CFR 50.59 review process. The second change corrects a typographical error that occurred under Amendment No. 176. The wording as proposed in this correction restores the requirement to the phraseology approved in Amendment No. 152 and is consistent with existing plant procedures.

Since there are no changes to the facility or facility procedures, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not modify the facility or the procedures for operation of the facility. One change updates the terminology used in 10 CFR 50.59 evaluations. The change does not alter the requirement of the TS Bases Control Program. The requirement for NRC review and approval of a TS Bases change is still determined through the use of the 10 CFR 50.59 review process. The second change corrects a typographical error that occurred under Amendment No. 176. The wording as proposed in this correction restores the requirement to the phraseology approved in Amendment No. 152 and is consistent with existing plant procedures.

Since there are no changes to the facility or facility procedures, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes continue to provide the controls necessary to ensure changes to the TS Bases are made in conformance with 10 CFR 50.59. The proposed changes continue to provide the controls necessary to ensure adequate control of High Radiation Areas. The proposed changes will not result in any changes to the facility or facility operating procedures. Therefore, the changes do not result in a significant reduction in the margin of safety.

Based on the above discussion, Carolina Power & Light has determined that the requested change does not involve a significant hazards consideration.

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

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

NRC Section Chief: Michael L. Marshall, Acting.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, Washington

Date of amendment request: June 9, 2004.

Description of amendment request: The proposed change revises Technical Specifications (TS) Limiting Condition for Operation (LCO) 3.4.11, ''RCS Pressure and Temperature (P/T) Limits,'' to replace the P/T curves for inservice leak and hydrostatic testing, non-nuclear heating and cooldown, and nuclear heating and cooldown currently illustrated in TS Figures 3.4.11–1, 3.4.11–2, and 3.4.11–3, respectively.

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

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

The proposed changes deal exclusively with the Reactor Coolant System (RCS) Pressure and Temperature (P/T) curves, which define the limitations for operation and testing. Because of the design conservatisms used to calculate the RCS P/T limits, reactor vessel failure has a low probability of occurrence and is not considered as a design basis accident in the safety analyses of the plant. The proposed changes adjust the reference temperature for the limiting material to account for irradiation effects and provide a comparable level of protection as previously evaluated and approved. The adjusted reference temperature calculations were performed in accordance with the requirements of 10 CFR [Part] 50 Appendix G using the guidance contained in RG [Regulatory Guide] 1.99, Revision 2, ''Radiation Embrittlement of Reactor Vessel Materials,'' to provide operating limits for up to 33.1 EFPY [effective full power years]. The proposed license amendment does not involve a change to operation of equipment required to mitigate any accident analyzed in Columbia's UFSAR [Updated Final Safety Analysis Report]. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The revised P/T curves are based on a later edition and addenda of the ASME Code that incorporates current industry standards for the curves. The revised curves are also based on an RPV [reactor pressure vessel] fluence that has been recalculated in accordance with the methodology of RG 1.190. The proposed changes do not involve a modification to plant equipment. There is no effect on the function of any plant system, and no new system interactions are introduced by this change. No new failure modes are introduced. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed curves conform to the guidance contained in RG 1.190, ''Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,'' and RG 1.99, Revision 2, ''Radiation Embrittlement of Reactor Vessel Materials,'' and maintain the safety margins specified in 10 CFR [Part] 50 Appendix G. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, Washington

Date of amendment request: August 5, 2004.

Description of amendment request: The proposed change will revise Technical Specification (TS) 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' to allow a one-time deferral of the Type A containment integrated leak rate test (ILRT). The current 10-year interval between Type A tests would be extended to 15 years from the previous time a Type A test was performed. The last Type A test was performed on July 20, 1994.

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

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

The proposed one-time extension to the Type A testing interval from once-per-10 years to once-per-15 years will not increase the probability of an accident previously evaluated. The performance of Type A tests is not an accident initiator. The primary containment Type A testing interval extension does not involve a plant modification and will not cause equipment failure or accident initiation.

The proposed extension to the Type A testing interval does not involve a significant increase in the consequences of an accident. The NUREG 1493 generic study of the effects of extending containment leakage testing concluded that Type B and C testing can identify the vast majority (greater than 95 percent) of potential leakage paths and that reducing the Type A test interval to once-per-20 years leads to an ''imperceptible increase in risk.'' Other testing and inspection programs, in addition to the Type A test, provide a high degree of assurance that the primary containment integrity will be maintained. Inspections required by the Maintenance Rule and ASME Code [are] periodically performed in order to identify indications of containment degradation that could affect containment leak tightness.

Experience at Columbia demonstrates that excessive containment leakage paths are detectable by Type B and C local leak rate tests. Type B and C testing will identify containment openings, such as a valve, that would otherwise be detected by the Type A test. These factors show that a one-time Type A test interval extension from once-per-10 years to once-per-15 years will not involve a significant increase in the consequences of an accident.

Previous Type A test results at Columbia show leakage has not exceeded acceptance criteria in the past, indicating a leak-tight containment and demonstrating the structural capability of the primary containment. The testing results have established that Columbia has had acceptable containment leakage rates with considerable margin.

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

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

The Columbia primary containment is designed to contain energy and fission products during and after a design basis accident. The proposed extension of the Type A testing interval will not create the possibility of a new or different type of accident from any previously evaluated. There are no changes being made to the physical plant or in operation of the plant that could introduce a new failure mode with the potential to create an accident or affect mitigation of an accident.

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

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

The proposed extension of the Type A testing interval will not significantly reduce the margin of safety. The NUREG 1493 generic study of the effects of extending containment leakage testing found that a 20-year interval in Type A leakage testing leads to an ''imperceptible increase in risk.'' NUREG 1493 found that generically, the design containment leakage rate contributes less than 0.1 percent to the overall accident risk and that the increase in the Type A testing interval would have a minimal effect on risk because the vast majority (greater than 95 percent) of all potential leakage paths are detected by Type B and C leakage testing.

A Columbia plant specific probabilistic risk assessment on the change in the Type A test interval from once-per-10 years to once-per-15 years determined:

- The risk impact due to a change in Large Early Release Frequency (LERF) is an increase of 2E–8/year that is characterized by Regulatory Guide 1.174 [''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis''] as ''very small.''
- The total integrated plant risk increase measured by person-rem/year is negligible.
- The change in conditional containment failure probability is an increase of 0.1 percent, which is considered to represent a very small impact on risk.

Deferral of Type A testing for Columbia does not increase the level of risk to the public due to loss of capability to detect and measure containment leakage or loss of containment structural integrity. Other containment testing methods and inspections will assure all limiting conditions for operation will continue to be met. The margin of safety inherent in existing accident analyses will be maintained.

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: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: June 22, 2004.

Description of amendment request: The proposed amendment would delete requirements from the Technical Specifications (TSs) to maintain hydrogen and oxygen monitors. A notice of availability for this technical specification improvement using the consolidated line item improvement process (CLIIP) was published in the **Federal Register** (FR) on September 25, 2003 (68 FR 55416). Licensees were generally required to implement upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to combustible gas control were imposed by Order for many facilities and were added to or included in the TSs for nuclear power reactors currently licensed to operate. The revised 10 CFR 50.44, ''Standards for combustible gas control system in light-water-cooled power reactors,'' eliminated the requirements for hydrogen recombiners (not installed at FitzPatrick and therefore not addressed by this proposed amendment) and relaxed safety classifications and licensee commitments to certain design and qualification criteria for hydrogen and oxygen monitors.

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the FR on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated June 22, 2004.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

NRC Section Chief: Richard J. Laufer.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: June 2, 2004.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to fully adopt the alternate source term (AST) methodology for design-basis accident dose consequence evaluations in accordance with 10 CFR 50.67. Specifically, the amendment would revise the TS Definition regarding dose equivalent iodine and TS Section 5.5.10, ''Ventilation Filter Testing Program (VFTP).'' The AST methodology for the fuel-handling accident was previously approved in Amendment No. 215, dated March 17, 2003.

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 involves the reanalysis of design basis radiological accidents in Containment and the Fuel Storage Building. The new analyses, based on the Alternate Source Term (AST), in accordance with 10 CFR 50.67, will replace the existing analyses that are based on the methodologies of [Atomic Energy Commission Report, ''Calculation of Distance Factors for Power and Test Reactor Sites,'' 1962] TID–14844. As a result of the new analyses, changes to the Technical Specifications are proposed which take credit for the new analysis results.

The proposed changes to the Technical Specifications modify requirements regarding filter testing for a variety of systems (*i.e.*, Containment Purge, Fuel Storage Building Emergency Ventilation). The analyses do not credit charcoal or HEPA [high-efficiency particulate air] filtration for dose mitigation. The proposed changes reflect the plant configuration that will support implementation of the AST analyses.

The AST analysis follows the guidance of the NRC Regulatory Guide 1.183 and uses the acceptance criteria of the NRC Standard Review Plan (NUREG–0800) for offsite doses and General Design Criteria for Control Room personnel. The accident analyses conservatively assume that the Containment Building and the Fuel Storage Building, including ventilation filtration systems for those buildings, do not diminish or delay the assumed fission product release.

The proposed changes also revise the definition of Dose Equivalent Iodine (DEI) to be consistent with the assumptions of the analyses. The limits for DEI do not change as a result of the implementation of the AST analyses.

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

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

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

Response: No.

The changes proposed in this license amendment request involve the use of a new analysis methodology and related regulatory acceptance criteria. In addition, certain changes to plant ventilation systems can be made based on the analysis results, using the new methodology. Use of a new analysis method does not impact the design or operation of plant systems or components and new accident scenarios would therefore not be created. The proposed changes to air ventilation and filtration systems do not adversely affect plant equipment used to protect plant safety limits or the way in which that plant equipment is operated or maintained. As a result, no new failure modes are being introduced that could lead to different accidents.

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

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

Response: No.

The existing dose analysis methodology and assumptions demonstrate that the dose consequences for all design basis accidents are within regulatory limits for whole body and thyroid doses as established in 10 CFR 100 (except for the Fuel Handling Analysis, which is already based on the AST methodology). The alternate dose analysis methodology and assumptions also demonstrate that the dose consequences of these accidents are within the regulatory requirements established for the new methodology.

The limits applicable to the alternate analysis are established in 10 CFR 50.67 in conjunction with the Total Effective Dose Equivalent (TEDE) acceptance directed in Regulatory Guide 1.183. The acceptance criteria for both dose analysis methods have been developed for the purpose of evaluating design basis accidents to demonstrate adequate protection of public health and safety. An acceptable margin of safety is inherent in both types of acceptance criteria.

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

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

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

NRC Section Chief: Richard J. Laufer.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: June 3, 2004.

Description of amendment request: The proposed amendment would increase the maximum authorized reactor core power level from 3067.4 megawatt thermal (MWt) to 3216 MWt. This represents a nominal increase of 4.85% rated thermal power. The amendment would also revise the Technical Specifications (TSs) to relocate certain cycle-specific parameters to the Core Operating Limits Report (COLR) by adopting TS Task Force Traveler TSTF–339, ''Relocate Technical Specification Parameters to the COLR.'' In addition, the amendment would revise several allowable values in TS Table 3.3.1–1, ''Reactor Protection System (RPS) Instrumentation,'' and Table 3.3.2–1, ''Engineered Safety Feature Actuation System (ESFAS) Instrumentation.''

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 evaluations and analyses associated with this proposed change to core power level have demonstrated that all applicable acceptance criteria for plant systems, components, and analyses (including the Final Safety Analysis Report Chapter 14 safety analyses) will continue to be met for the proposed increase in licensed core thermal power for Indian Point 3 (IP3). The subject increase in core thermal power will not result in conditions that could adversely affect the integrity (material, design, and construction standards) or the operational performance of any potentially affected system, component or analysis. Therefore, the probability of an accident previously evaluated is not affected by this change. The subject increase in core thermal power will not adversely affect the ability of any safety-related system to meet its intended safety function. Further, the radiological dose evaluations in support of this power uprate effort show all acceptance criteria are met.

The relocation of cycle-specific core operating limits from the Technical Specifications to the Core Operating Limits Report (COLR), in accordance with TSTF–339, has no influence or impact on the probability or consequences of a Design Basis Accident. Adherence to the COLR and accepted methodologies for establishing COLR parameters continues to be controlled by the plant Technical Specifications. Relocation of cycle-specific values to the COLR while maintaining the limiting requirements in the Technical Specifications reduces administrative burden associated with processing license amendments for routine core reload designs.

RPS and ESF [engineered safety feature] allowable values established in plant technical specifications represent acceptance criteria used by plant personnel in assessing the operability of instrumentation channels.

Allowable values are not accident initiators and have no role in the probability of occurrence of an accident. Safety analyses for design basis accidents use certain assumptions (Safety Analysis Limits) regarding the actuation of RPS and ESF protective functions. The proposed allowable values are developed using a methodology that assures the accident analysis assumptions are valid and the consequences of previously analyzed accidents continue to meet established limits.

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

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

Response: No.

The analyses and evaluations performed for the proposed increase in power show that all applicable acceptance criteria for plant systems, components, and analyses (including FSAR [Final Safety Analysis Report] Chapter 14 safety analyses) will continue to be met for the proposed power increase in IP3 licensed core thermal power. The subject increase in core thermal power will not result in conditions that could adversely affect the integrity (material, design, and construction standards) or operational performance of any potentially affected system, component, or analyses. The subject increase in core thermal power will not adversely affect the ability of any safety-related system to meet its safety function. Furthermore, the conditions and changes associated with the subject increase in core thermal power will neither cause initiation of any accident, nor create any new credible limiting single failure. The power uprate does not result in changing the status of events previously deemed to be non-credible being made credible. Additionally, no new operating modes are proposed for the plant as a result of this requested change.

The relocation of cycle-specific core operating limits from the Technical Specifications to the Core Operating Limits Report (COLR), in accordance with TSTF–339, does not involve any changes to plant equipment or the way is which the plant is operated. There are no new accident initiators or causal mechanisms being introduced by this proposed change. Relocation of cycle-specific values to the COLR while maintaining the limiting requirements in the Technical Specifications reduces administrative burden associated with processing license amendments for routine core reload designs.

RPS and ESF allowable values established in plant technical specifications represent acceptance criteria used by plant personnel in assessing the operability of instrumentation channels. Revising allowable values does not involve installation of new equipment, modification to existing equipment, or a change in plant operation that could create a new or different accident scenario.

Therefore, the proposed changes described in this license amendment request 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 analyses and evaluations associated with the proposed increase in power show that all applicable acceptance criteria for plant systems, components, and analyses (including FSAR Chapter 14 safety analyses) will continue to be met for this proposed increase in IP3 licensed core thermal power. The subject increase in core thermal power will not result in conditions that could adversely affect the integrity (material, design, and construction standards) or operational performance of any potentially affected system, component, or analysis. The subject power uprate will not adversely affect the ability of any safety-related system to meet its intended safety function.

Adoption of TSTF–339 allows relocation of cycle-specific parameters to the COLR, while maintaining limiting requirements in the Technical Specifications. Approved methodologies for calculating cycle-specific parameters are maintained in the Technical Specifications, and changes to the COLR are subject to the requirements and controls of 10 CFR 50.59. This assures that required margins to safety limits are maintained.

The proposed new allowable values are developed using established methodologies and incorporate additional conservatism that assures the validity of analysis limits assumed in the evaluation of hypothetical accidents.

Therefore, the proposed changes described in this license amendment request will not involve a significant reduction in [a] margin of safety.

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

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: July 8, 2004.

Description of amendment request: Delete Technical Specification Surveillance Requirement 4.5.2.d.1, Emergency Core Cooling System Subsystems $-T_{ave} \geq 300$ °F, associated with the requirement to maintain an operable Automatic Closure Interlock (ACI) for the Shutdown Cooling (SDC) suction isolation valves.

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 removal of the ACI function is consistent with the guidelines previously endorsed by the NRC in Generic Letter 88–17. Removal of this function results in a calculated decrease in intersystem Loss of Coolant Accident (ISLOCA) frequency. Additionally, the removal of the ACI function will result in a decrease in SDC system unavailability and a corresponding decrease in risk associated with loss of SDC events. As a result, the proposed change will result in a net decrease in risk and a net improvement in plant safety.

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 presence or omission of an ACI function is not considered an accident initiator nor is this function credited in any safety analyses for the prevention or mitigation of any accident. Alarms, design features, and strict administrative/procedural controls support correct and timely operator action to ensure the SDC system will not be exposed to high Reactor Coolant System (RCS) pressure.

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

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

Response: No.

The ACI function is not credited in a margin of safety analysis for any accident previously evaluated. Removal of the ACI function will result in an overall net increase in nuclear safety. Appropriate alarm, design features, and administrative controls will continue to ensure proper isolation and isolation maintenance of the SDC system during plant operations with elevated RCS pressures.

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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: July 8, 2004. This supersedes the May 12, 2004, application in its entirety (69 FR 34699).

Description of amendment request: The proposed amendment would change the reactor core analytical methods used to determine the core operating limits, reflect the changes allowed by Technical Specification (TS) Task Force (TSTF) Traveler No. 363, ''Revised Topical Report References in ITS [Improved Standard Technical Specifications] 5.6.5, COLR [Core Operating Limits Report],'' and delete the Index from the TSs. This request completely supersedes the previous request of May 12, 2004.

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.

TS 6.9.5.1, Core Operating Limits Report (COLR)

The proposed amendment, in part, identifies a change in the nuclear physics codes used to confirm the values of selected cycle-specific reactor physics parameter limits and includes minor editorial changes which do not alter the intent of stated requirements. The proposed change also allows the use of methods required for the implementation of ZIRLO clad fuel rods. Inasmuch as the proposed change includes codes that have been previously approved by the NRC for CE [Combustion Engineering] cores, the amendment is administrative in nature and has no impact on any plant configuration or system performance relied upon to mitigate the consequences of an accident. Parameter limits specified in the COLR for this amendment are not changed from the values presently required by TSs. Future changes to the calculated values of such 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. Assumptions used for accident initiators and/or safety analysis acceptance criteria are not altered by this change.

The proposed change will add an NRC approved topical report, WCAP–16072–P–A, to the list of referenced topical reports. The topical report has been previously approved by the NRC for use in Combustion Engineering core designs and as such, the proposed change is administrative in nature and has no impact on any plant configurations or on system performance that is relied upon to mitigate the consequences of an accident. In addition, prior to the use of the ZrB_2 burnable absorber coating, fuel design will be analyzed with applicable NRC staff approved codes and methods.

The proposed change also implements NRC approved TSTF Traveler No. 363. This is an administrative change that will allow specific details, such as the revision number, revision date, and supplement number of topical reports that are referenced in the TSs, to be deleted and relocated in the cycle specific COLR. This proposed change does not result in any changes to the assumptions used to evaluated [evaluate] accident initiators and/or safety analysis acceptance criteria.

Index

The proposed deletion of the Index is purely administrative and does not impact the accident analysis.

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.

TS 6.9.5.1, Core Operating Limits Report (COLR)

The proposed change, in part, identifies a change in the nuclear physics codes used to confirm the values of selected cycle-specific reactor physics parameter limits. The proposed change also allows the use of methods required for the implementation of ZIRLO clad fuel rods. Neither of these changes results in a change to the physical plant or to the modes of operation defined in the facility license.

The proposed change adds a reference to the topical report that allows the use of ZrB_2 as a burnable absorber coating on the fuel pellet. The topical report has been previously approved by the NRC for use in Combustion Engineering core designs and as such, the proposed change is administrative in nature and has no impact on any plant configurations or on system performance that is relied upon to mitigate the consequences of an accident. In addition, prior to the use of the ZrB_2 burnable absorber coating, fuel design will be analyzed with applicable NRC staff approved codes and methods. This change is administrative in nature and does not create a new or different type of accident than previously evaluated because the design requirements for the facility remain the same.

The proposed change also implements TSTF Traveler No. 363. The proposed change does not result in changes to the physical plant or to the modes of operation defined in the facility license nor does it involve the addition of new equipment or the modification of existing equipment.

Index

The proposed deletion of the Index is purely administrative has no affect on existing equipment.

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.

TS 6.9.5.1, Core Operating Limits Report (COLR)

The proposed changes to change the nuclear physics code package and to add a topical report to support the use of ZIRLO do not amend the cycle specific parameter limits located in the COLR from the values presently required by the TS. The individual specifications continue to require operation of the plant within the bounds of the limits specified in COLR. Benchmarking has shown that uncertainties for the Westinghouse Physics code system yields are essentially the same or less than those obtained for the current ROCS and DIT [computer code] methodology. Future changes to the values of these limits by the licensee may only be developed using NRC approved methodologies, must remain consistent with all applicable plant safety analysis limits addressed in the Safety Analysis Report, and are further controlled by the 10 CFR 50.59 process. The relocation of the supplement numbers, revision numbers, and approval dates of the analytical methods listed in the COLR does not affect the margin of safety. The analysis will continue to be performed using NRC approved methodology. Safety analysis acceptance criteria are not being altered by this amendment.

The proposed change will add WCAP–16072–P–A to the list of referenced topical reports. The topical report has been previously approved by the NRC for use in Combustion Engineering core designs and as such, the proposed change is administrative in nature and has no impact on any plant configurations or on system performance that is relied upon to mitigate the consequences of an accident. In addition, prior to the use of the ZrB_2 burnable absorber coating, fuel design will be analyzed with applicable NRC staff approved codes and methods.

Index

The proposed deletion of the Index, which is an administrative document, does not impact any TS values or safety 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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: June 10, 2004, as supplemented by letter dated July 21, 2004.

Description of amendment request: The proposed amendments would revise the Quad Cities Nuclear Power Station (QCNPS) technical specifications (TS) to change the allowable value (AV) and add surveillance requirements (SRs) for the main steam line (MSL) flow-high initiation of Group 1 primary containment isolation and control room emergency ventilation system isolation.

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

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

For QCNPS, Units 1 and 2, the proposed amendment will implement a design change that upgrades the existing MSL Flow-High instrumentation from pressure switches to analog trip unit devices. Analog trip units (ATUs) have proven to be a more reliable technology than the currently installed equipment. Analog trip units are used in various applications at QCNPS, including the Reactor Protection System (RPS) low water level trip function. Because the trip units are more reliable, the likelihood of spurious isolations is reduced. Further, ATUs experience less instrument drift during the operating cycle. The proposed change adds a 92-day trip unit calibration requirement for the MSL-High isolation function. The NRC has previously found that a 92-day calibration is appropriate for individual ATUs.

Procedure revisions required by this modification are limited to those associated with the calibration, maintenance, and operation of the replacement transmitter and trip unit analog loops. All required design functions of the MSL high flow loop are maintained. No system, structure, or component will be used in a manner that is not already bounded by the reference design, or is inconsistent with analyses or descriptions in the QCNPS Updated Final Safety Analysis Report (UFSAR). There is no adverse effect on the performance or control of any design function described in the UFSAR.

TS requirements that govern operability or routine testing of plant instruments are not assumed to be initiators of any analyzed event because these instruments are intended to prevent, detect, or mitigate accidents. Therefore, these changes will not involve an increase in the probability of occurrence of an accident previously evaluated. In addition, these changes will not increase the consequences of an accident previously evaluated because the proposed change does not adversely impact structures, systems, or components. The planned instrument upgrade is a more reliable design than existing equipment. The proposed changes establish requirements that ensure components are operable when necessary for the prevention or mitigation of accidents or transients. Furthermore, there will be no change in the types or significant increase in the amounts of any effluents released offsite. For these reasons, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes support a planned instrumentation upgrade by incorporating SRs required to ensure operability. The change does not adversely impact the manner in which the instrument will operate under normal and abnormal operating conditions. Therefore, these changes provide an equivalent level of safety and will not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions.

All required design functions are maintained, and the new setpoint is analyzed in accordance [with] an NRC-approved methodology for determination of setpoints and TS AVs in accordance with the QCNPS UFSAR, Section 7.3.2.4, ''Design Evaluation.'' Therefore, replacing the existing MSL high flow DPISs with analog trip instrumentation does not alter any UFSAR described evaluation methodologies, or introduce any new methodologies. These changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes support a planned instrumentation upgrade from differential pressure switches to ATUs. The proposed changes do not adversely affect the probability of failure or availability of the affected instrumentation. The addition of a 92-day trip unit calibration for MSL Flow-High is a conservative change that aligns the SRs for a planned instrumentation upgrade with that of similar instrumentation. The NRC has previously found that a 92-day calibration is appropriate for individual ATUs. The setpoint was determined using an NRC-approved methodology. The proposed changes do not affect the analytical limit assumed in the safety analyses for the actuation of the instrumentation. Therefore, it is concluded that the proposed changes will not result in a 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. Thomas S. O'Neill, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS–1 and 2), Beaver County, Pennsylvania

Date of amendment request: March 22, 2004 as supplemented July 23, 2004.

Description of amendment request: The proposed change allows entry into a mode or other specified condition in the applicability of a technical specification (TS), while in a condition statement and the associated required actions of the TS, provided the licensee performs a risk assessment and manages risk consistent with the program in place for complying with the requirements of Title 10 of the Code of Federal Regulations (10 CFR), part 50, Section 50.65(a)(4). Limiting Condition for Operation (LCO) 3.0.4 exceptions in individual TSs would be eliminated, several notes or specific exceptions are revised to reflect the related changes to LCO 3.0.4, and Surveillance Requirement (SR) 4.0.4 is revised to reflect the LCO 3.0.4 allowance.

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–359. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 2, 2002 (67 FR 50475), on possible amendments concerning TSTF–359, 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 April 4, 2003 (68 FR 16579). The licensee affirmed the applicability of the following NSHC determination in its application dated March 22, 2004 and July 23, 2004, supplement.

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 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. Being in a TS condition and the associated required actions is 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 this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Therefore, this 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 Previously Evaluated

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). 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 this change will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change allows 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 TS allow operation of the plant without the full complement of equipment through the 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 condition on the margin of safety is not considered significant. The proposed change does not alter the required actions or completion times of the TS. The proposed change allows 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 change also eliminates 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, this 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: Mary O'Reilly, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Richard J. Laufer.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit No. 2 (BVPS–2), Beaver County, Pennsylvania

Date of amendment request: July 23, 2004.

Description of amendment request: The proposed amendment would revise the BVPS–2 Technical Specifications to eliminate periodic response time testing requirements on selected sensors and selected protection channel components and permit the option of measuring or verifying the response times by means other than testing.

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.

This change to the Technical Specifications does not result in a condition where the design, material, and construction standards that were applicable prior to the change are altered. The same RTS [reactor trip system] and ESFAS [engineered safety features actuation system] instrumentation is being used; the time response allocations/modeling assumptions in the Updated Final Safety Analysis Report (UFSAR) Chapter 15 analyses are still the same; only the method of verifying [the] time response is changed. The proposed change will not modify any system interface and could not increase the likelihood of an accident since these events are independent of this change. The proposed activity will not change, degrade or prevent actions or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the UFSAR.

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.

This change does not alter the performance of the pressure and differential pressure transmitters, process protection racks, Nuclear Instrumentation, and logic systems used in the Reactor Trip and Engineered Safety Features Actuation Systems. All sensors, process protection racks, Nuclear Instrumentation, and logic systems will still have response time verified by [a] test before placing the equipment into operational service and after any maintenance that could affect the response time. Changing the method of periodically verifying instrument response times for certain equipment (assuring equipment operability) from time response testing to calibration and channel checks will not create any new accident initiators or scenarios. Periodic surveillance of these instruments will detect significant degradation in the equipment response time characteristics.

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.

This change does not affect the total system response time assumed in the safety analysis. The periodic system response time verification method for selected sensors and differential pressure sensors and for process protection racks, Nuclear Instrumentation, and logic systems is modified to allow use of actual test data or engineering data. The method of verification still provides assurance that the total system response time is within that assumed in the safety analysis.

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

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

Attorney for licensee: Mary O'Reilly, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: April 26, 2004.

Description of amendment request: This proposed license amendment would revise the frequency of the Mode 5 Intermediate Range Monitoring (IRM) Instrumentation CHANNEL FUNCTIONAL TEST contained in Technical Specification (TS) 3.3.1.1 from 7 days to 31 days. The methodology used to analyze the change in testing frequency is based upon guidance contained in Generic Letter 91–04, ''Changes in Technical Specification Surveillance Intervals to Accommodate a 24-month Fuel Cycle,'' and Electric Power Institute (EPRI) Report TI–103335, ''Guidance for Instrumentation Calibration Extension/Reduction Programs.''

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

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

The proposed Technical Specification (TS) change involves an increase in the Mode 5 CHANNEL FUNCTIONAL TEST interval for Reactor Protection System (RPS) Intermediate Range Monitor (IRM) from 7 days to 31 days. The proposed TS change does not alter the design or functional requirements of the RPS or IRM systems. Evaluation of the proposed testing interval change demonstrated that the availability of the IRMs to prevent or mitigate the consequences of a control rod withdrawal event at low power levels are not significantly affected because of other, more frequent testing that is performed, the availability of redundant systems and equipment, and the high reliability of the IRM equipment.

Furthermore, using the guidance of GL 91–04, a historical review of surveillance test results and associated maintenance records did not indicate evidence of any failure that would invalidate the above conclusions.

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

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

The proposed TS change involves an increase in the Mode 5 IRM CHANNEL FUNCTIONAL TEST interval from 7 days to 31 days. Existing TS testing requirements ensure the operability of the IRMs. The proposed TS change does not introduce any failure mechanisms of a different type than those previously evaluated, since no physical changes to the plant are being made. No new or different equipment is being installed, and no installed equipment is being operated in a different manner. As a result, no new failure modes are introduced. In addition, the manner in which surveillance tests are performed remain unchanged.

Furthermore, using the guidance in GL 91–04, a historical review of surveillance test results and associated maintenance records did not indicate evidence of any failure that would invalidate the above conclusions.

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

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

The proposed Technical Specifications (TS) change involves an increase in the Mode 5 CHANNEL FUNCTIONAL TEST interval for Reactor Protection System (RPS) Intermediate Range Monitor (IRM) from 7 days to 31 days. The impact on system operability is minimal, based upon performance of the more frequent Channel

Checks, continuous Control Room monitoring when the IRMs are in use, and the overall IRM reliability. Evaluations show there is no evidence of time-dependent failures that would impact the availability of the IRMs.

Furthermore, using the guidance in GL 91–04, a historical review of surveillance test results and associated maintenance records did not indicate evidence of any failure that would invalidate the above conclusions.

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: Mary E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: June 28, 2004.

Description of amendment request: The proposed amendment would revise Technical Specification 3/4.9.4, ''Containment Building Penetrations,'' to align the language of the Surveillance Requirement with the Applicability Statement contained in the Limiting Condition for Operation.

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

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

The proposed change aligns the language of the Surveillance Requirement for Containment Building Penetrations with the language of the Applicability Statement of Technical Specification 3.9.4.

The proposed amendment will not change the design function, or method of performing or controlling design functions, of structures, systems and components, nor will there be an effect on FPL Energy Seabrook programs. As a result, the proposed amendment will not change assumptions, or change, degrade or prevent actions described or assumed in accidents evaluated and described in the Seabrook Station UFSAR [updated final safety analysis report]. The proposed change to the Surveillance Requirement wording does not adversely affect performance of the Surveillance Requirement that verifies the status of Containment Building Penetrations. Since the status of the Containment Penetrations is not adversely affected by the proposed change, the radiological consequences of an event are unchanged. Therefore, the proposed amendment does not result in an increase in the radiological consequences of any accident described in the Seabrook Station UFSAR.

Therefore, it is concluded that these proposed changes do not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

The proposed change aligns the language of the Surveillance Requirements for Containment Building Penetrations with the language in the Applicability Statement of the Technical Specification.

The proposed amendment will not change the design function, or method of performing or controlling design functions, of structures, systems and components, nor will there be an effect on FPL Energy Seabrook programs. As a result, there are no changes associated with the proposed amendment that could potentially introduce new failure modes or accident scenarios.

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

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

The proposed change aligns the language of the Surveillance Requirement for Containment Building Penetrations with the language of the Applicability Statement of Technical Specification 3.9.4. The proposed amendment does not change the design function, or method of performing or controlling design functions, of structures, systems and components, nor will there be an effect on FPL Energy Seabrook programs. The status of containment penetrations will continue to be verified. The proposed change does not involve any changes to a margin of safety.

Therefore, it is concluded that these proposed changes do not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: August 17, 2004.

Description of amendment request: The licensee proposed to revise Section 3.3.1, ''Oxygen Concentration [of the primary containment],'' of the Technical Specifications (TSs) to (1) add a new action allowing 24 hours to restore the oxygen concentration to within the limit of <4% by volume if the limit is exceeded when the reactor is in the power operating condition, and (2) incorporate the associated conforming changes of editorial nature. The proposed 24-hour completion time for restoring oxygen concentration is consistent with Improved Standard Technical Specifications for Boiling Water Reactors (NUREG–1433, Revision 3).

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

The first standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The effect of the proposed amendment is to provide the same 24-hour completion time to restore oxygen concentration to under the 4% limit should the oxygen concentration rise due to other than a reactor shutdown-startup evolution. The proposed amendment does not lead to, nor is it the result of, a plant design change. These TS changes will not lead to alteration of the physical design or operational procedures associated with the containment system, or any other plant structure, system, or component (SSC). All requirements needed to assure operability of the containment system will remain unchanged. Containment atmospheric oxygen concentration was not assumed to be a precursor of accidents, nor was it assumed to be a component in previously evaluated accident scenarios. Accordingly, the revised specifications will lead to no increase in the consequences of an accident previously evaluated, and no increase of the probability of an accident previously evaluated.

The second standard requires that operation of the unit in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. As stated above, the proposed amendment involves only the time allowed to restore containment atmospheric oxygen concentration to under 4 percent by volume, and associated editorial changes. These changes do not alter the physical design, safety limits, or method of operation associated with the operation of the plant. Accordingly, the changes do not introduce any new or different kind of accident from those previously evaluated.

The third standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant reduction in a margin of safety. Since the licensee did not propose to exceed or alter a design basis or safety limit, did not propose to operate any component in a less conservative manner, and did not propose to use a less conservative analysis methodology, the proposed amendment will not affect in any way the performance characteristics and intended functions of any SSC. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: July 6, 2004.

Description of amendment request: The proposed change involves the extension from 1 hour to 24 hours for the completion time (CT) of Technical Specification (TS) 3.3.a.2.B, which defines requirements for accumulators. Accumulators are part of the emergency core cooling system and consist of tanks partially filled with borated water and pressurized with nitrogen gas. The contents of the tank are discharged to the reactor coolant system (RCS) if, as during a loss-of-coolant accident, the coolant pressure decreases to below the accumulator pressure. TS 3.3.a.2.B specifies a CT to restore an accumulator to operable status when it has been declared inoperable for a reason other than the boron concentration of the water in the accumulator not being within the required range. This change was proposed by the Westinghouse Owners Group participants in the TS Task Force (TSTF) and is designated TSTF–370, ''Increase Accumulator Completion Time from 1 Hour to 24 Hours.'' TSTF–370 is supported by NRC-approved Topical Report WCAP–15049–A, ''Risk-Informed Evaluation of an Extension to Accumulator Completion Times,'' submitted on May 18, 1999. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on July 15, 2002 (67 FR 46542), on possible amendments concerning TSTF–370, 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 March 12, 2003 (68 FR 11880). The licensee affirmed the applicability of the following NSHC determination in its application dated July 6, 2004.

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 basis for the accumulator limiting condition for operation (LCO), as discussed in [Standard Technical Specifications] Bases Section 3.5.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of the WCAP–15049, ''Risk-Informed Evaluation of an Extension to Accumulator Completion Times,'' evaluation, the proposed change will allow plant operation in a configuration outside the design basis for up to 24 hours, instead of 1 hour, before being required to begin shutdown. The impact of the increase in the accumulator CT on core damage frequency for all the cases evaluated in WCAP–15049 is within the acceptance limit of 1.0E–06/yr for a total plant core damage frequency (CDF) less than 1.0E–03/yr. The incremental conditional core damage probabilities calculated in WCAP–15049 for the accumulator CT increase meet the criterion of 5E–07 in Regulatory Guides (RG) 1.174 and 1.177 for all cases except those that are based on design basis success criteria. As indicated in WCAP–15049, design basis accumulator success criteria are not considered necessary to mitigate large break loss-of-coolant accident (LOCA) events, and were only included in the WCAP–15049 evaluation as a worst case data point. In addition, WCAP–15049 states that the NRC has indicated that an incremental conditional core damage frequency (ICCDP) greater than 5E–07 does not necessarily mean the change is unacceptable. The proposed technical specification change does not involve any hardware changes nor does it affect the probability of any event initiators. There will be no change to normal plant operating parameters, engineered safety feature (ESF) actuation setpoints, accident mitigation capabilities, accident analysis assumptions or inputs. Therefore, this 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 Previously Evaluated

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. As described in Section 9.1 of the WCAP–15049 evaluation, the plant design will not be changed with this proposed technical specification CT increase. All safety systems still function in the same manner and there is no additional reliance on additional systems or procedures. The proposed accumulator CT increase has a very small impact on core damage frequency. The WCAP–15049 evaluation demonstrates that the small increase in risk due to increasing the accumulator allowed outage time (AOT) is within the acceptance criteria provided in RGs 1.174 and 1.177. No new accidents or transients can be introduced with the requested change and the likelihood of an accident or transient is not impacted. The malfunction of safety related equipment, assumed to be operable in the accident analyses, would not be caused as a result of the proposed technical specification change. No new failure mode has been created and no new equipment performance burdens are imposed. 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 involve a significant reduction in a margin of safety. There will be no change to the departure from nucleate boiling ratio (DNBR) correlation limit, the design DNBR limits, or the safety analysis DNBR limits. The basis for the accumulator LCO, as discussed in Bases Section 3.5.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of the WCAP–15049 evaluation, the proposed change will allow plant operation in a configuration outside the design basis for up to 24 hours, instead of 1 hour, before being required to begin shutdown. The impact of this on plant risk was evaluated and found to be very small. That is, increasing the time the accumulators will be unavailable to respond to a large LOCA event, assuming accumulators are needed to mitigate the design basis event, has a very small impact on plant risk. Since the frequency of a design basis large LOCA (a large LOCA with loss of offsite power) would be significantly lower than the large LOCA frequency of the WCAP–15049 evaluation, the impact of increasing the accumulator CT from 1 hour to 24 hours on plant risk due to a design basis large LOCA would be significantly less than the plant risk increase presented in the WCAP–15049 evaluation. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the reasoning presented above and the previous discussion of the amendment request, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: July 6, 2004.

Description of amendment request: The proposed amendment relocates the surveillance requirements for Item 22, ''Accumulator Level and Pressure,'' and Item 25, ''Portable Radiation Survey Instruments,'' from Table TS 4.1–1 of the Technical Specifications to licensee-controlled documents.

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?

NMC [Nuclear Management Company] Response for Proposed Change to Table TS 4.1–1, Item 22

No. This TS change removes the accumulator water level and pressure channel surveillance from the TS and places them into licensee controlled documents. This change is consistent with industry and NRC [Nuclear Regulatory Commission] recognition that the accumulator instrumentation operability is not directly related to the capability of the accumulators to perform their safety function.

Relocating the instrumentation surveillance requirements is an administrative change that will not affect equipment testing, availability, or operation. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

NMC Response for Proposed Change to Table TS 4.1–1, Item 25

No. Removing the surveillance requirements for portable radiation survey instruments from the TS is administrative and has no impact on plant equipment, accident initiators, or the safety analysis. Additionally, eliminating the monthly check and modifying the line item description does not impact plant equipment or operation. Therefore, the change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

NMC Response for Proposed Change to Table TS 4.1–1, Item 22

No. Relocating the accumulator water level and pressure instrument surveillance requirements to licensee controlled documents is an administrative change that will not change any equipment, require new equipment to be installed, or change the way current equipment operates in the plant.

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

NMC Response for Proposed Change to Table TS 4.1–1, Item 25

No. Removing the surveillance requirements for portable radiation survey instruments from the TS and relocating the requirements to licensee controlled documents is administrative and has no impact on plant equipment or the way the plant equipment operates. Additionally, eliminating the monthly check and modifying the line item description does not impact plant equipment or operation. Portable radiation survey instruments are not accident initiators. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

NMC Response for Proposed Change to Table TS 4.1–1, Item 22

No. Relocating the accumulator water level and pressure instrument surveillance requirements to licensee controlled documents is an administrative change that will not change the safety analyses performed for the plant nor reduce the ability of the accumulators to perform their safety related function. There is no change in the operation of the accumulators or related equipment and systems. Therefore, the change does not involve a reduction in the margin of safety.

NMC Response for Proposed Change to Table TS 4.1–1, Item 25

No. Portable radiation survey instruments are not inputs to the safety analysis or to automatic plant actions. The change is administrative since it moves the requirements out of TS and into licensee controlled documents through use of the 10 CFR 50.36 selection criteria for TS. Additionally, eliminating the monthly check and modifying the line item description does not impact plant equipment or operation. Therefore, the change does not reduce the margin of safety.

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

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: August 2, 2004.

Description of amendment request: The proposed amendment would implement a risk-informed process for determining allowed outage times for South Texas Project (STP), Units 1 and 2, Technical Specifications (TS). The risk-informed process involves the application of the STP, Units 1 and 2, Configuration Risk Management Program (CRMP). The STP CRMP is a procedurally controlled program utilized for the implementation of 50.65(a)(4) of Title 10 of the Code of Federal Regulations (10 CFR).

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 to the Technical Specifications involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed changes to the Technical Specifications to add a new TS 3.13.1 and to change specific TS to apply the new TS 3.13.1 do not involve a significant increase in the probability of an accident previously evaluated because the changes involve no change to the plant or its modes of operation. In addition, the risk-informed configuration management program will be applied to effectively manage the availability of required systems, structures, and components to assure there is no significant increase in the probability of an accident. These proposed changes do not increase the consequences of an accident because the design-basis mitigation function of the affected systems is not changed and the risk-informed configuration management program will be applied to effectively manage the availability of systems, structures and components required to mitigate the consequences of an accident. The application of the risk-informed configuration management program is considered a substantial technological improvement over current methods.

Therefore, none of the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated.

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

None of the proposed changes involve a new mode of operation or design configuration. There are no new or different systems, structures, or components proposed by these changes. Therefore, there is no possibility of a new or different kind of accident.

3. Does the proposed change to the Technical Specifications involve a significant reduction to a margin of safety?

Proposed new TS 3.13.1 and the associated changes to the specifications that apply the new TS 3.13.1 implement a risk-informed configuration management program to assure that adequate margins of safety are maintained. Application of these new specifications and the configuration management program considers cumulative effects of multiple systems or components being out of service and does so more effectively than the current Technical Specifications. Therefore, application of these new specifications will not involve a significant reduction in a margin of safety.

Based on the evaluation above, none of the proposed changes 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 standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: August 12, 2004.

Description of amendment request: The proposed changes to the South Texas Project (STP), Units 1 and 2, Technical Specifications (TS) for steam generators (SGs) are based on draft TS Task Force (TSTF) Improved Standard TS Change Traveler TSTF–449, Rev. 2, and the Joseph M. Farley Nuclear Plant, Units 1 and 2, submittal dated June 28, 2004, as supplemented by letter dated August 5, 2004. The changes would implement guidance for the industry initiative on Nuclear Energy Institute (NEI) 97–06, ''Steam Generator Program Guidelines.''

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 requires a Steam Generator Program that includes performance criteria that will provide reasonable assurance that the SG tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown, and all anticipated transients included in the design specification). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational leakage.

The structural integrity performance criterion is:

All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown, and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 (3 [delta] P) against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

The accident induced leakage performance criterion is:

The primary-to-secondary accident induced leakage rate for any design basis accidents, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Accident induced leakage is not to exceed 1 gpm [gallons per minute] total for all four SGs in a unit.

The operational leakage performance criterion is:

''The RCS operational primary-to-secondary leakage through any one SG shall be limited to 150 gallons per day.''

An SGTR [steam generator tube rupture] event is one of the design basis accidents analyzed as part of the plant licensing basis. In the analysis of an SGTR event, a bounding primary-to-secondary leakage rate equal to the operational leakage rate limits in the licensing basis plus the leakage rate associated with a double-ended rupture of a single tube is assumed.

For other design basis accidents such as MSLB [main steamline break], rod ejection, and reactor coolant pump locked rotor, the tubes are assumed to retain their structural integrity (*i.e.*, they are assumed not to rupture). At STP these analyses assume that the total primary-to-secondary leakage is 1 gpm. The accident induced leakage criterion introduced by the proposed changes accounts for tubes that may leak during design basis accidents. The accident induced leakage criterion limits this leakage to no more than the value assumed in the accident analysis.

The SG performance criteria proposed in this change to the TS identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining RCPB [reactor coolant pressure boundary] integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the Steam Generator Program required by the proposed change to the TS. The program, defined by NEI 97–06, includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring.

The consequences of design basis accidents are, in part, functions of the dose equivalent I–131 in the primary coolant and the primary-to-secondary leakage rates resulting from an accident. Therefore, limits are included in the TS for operational leakage and for dose equivalent I–131 in primary coolant to ensure the plant is operated within its analyzed condition. The analysis of the limiting design basis accident assumes that primary-to-secondary leak rate after the accident is 1 gpm with no more than 500 gpd [gallons per day] in any one SG, and that the reactor coolant activity levels of dose equivalent I–131 are at the TS values before the accident.

The proposed change does not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed approach updates the current TS and enhances the requirements for SG inspections. The proposed change does not adversely impact any other previously evaluated design basis accident and is an improvement over the current TS.

Therefore, the proposed change does not affect the consequences of an SGTR accident and the probability of such an accident is reduced. In addition, the proposed changes do not affect the consequences of an MSLB, rod ejection, or a reactor coolant pump locked rotor event.

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 performance-based requirements are an improvement over the requirements imposed by the current TS.

Implementation of the proposed Steam Generator Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The result of the implementation of the Steam Generator Program will be an enhancement of SG tube performance. Primary-to-secondary leakage that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. In addition, the proposed change does not impact any other plant system or component. The change enhances SG inspection requirements.

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

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

Response: No.

The SG tubes are an integral part of the RCPB and, as such, are relied upon to maintain the primary system pressure and inventory. As part of the RCPB, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a SG is maintained by ensuring the integrity of its tubes.

Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in tube integrity by implementing the Steam Generator Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the Steam Generator Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TS.

For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by the proposed change to the TS.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: August 5, 2004.

Brief description of amendments: The proposed change revises Technical Specification 3.7.10 entitled, ''Control Room Emergency Filtration/Pressurization System (CREFS),'' to add a new condition for an inoperable Control Room boundary.

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

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

Response: No.

This is a revision to the Technical Specifications for the Control Room Emergency/Filtration System which is a mitigation system designed to minimize in leakage and to filter the control room atmosphere to protect the operator following accidents previously analyzed. An important part of the system is the Control Room boundary. The Control Room boundary integrity is not an initiator or precursor to any accident previously evaluated. Therefore, the probability of any accident previously evaluated is not increased. The analysis of the consequences of analyzed accident scenarios under the control room breach conditions along with the compensatory actions for restoration of control room integrity demonstrate that the consequences of any accident previously evaluated are not increased. Therefore, it is concluded that this change does not significantly increase the probability [or consequences] of an accident previously evaluated.

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

Response: No.

The proposed change will not impact the accident analysis. The change will not alter the requirements of the Control Room Emergency/Filtration System or its function during accident conditions. The administrative controls and compensatory actions will ensure the control room emergency/filtration system will perform its safety function. No new or different accidents result from performing the new actions and surveillance required. The change does not involve a physical alteration of the plant (*i.e.*, no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The change does not alter assumptions made in the safety analysis. The proposed change is consistent with the safety analysis assumptions and current plant operating practice. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by these changes. The proposed change will not result in plant operation in a configuration outside the design basis for an unacceptable period of time without compensatory actions and administrative controls. The proposed change does not affect systems that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition. Therefore the proposed change does not involve a 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: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: July 1, 2004.

Description of amendment request: The proposed license amendments would modify the Reactor Coolant System (RCS) pressure/temperature (P/T) limit curves, the Low-Temperature Overpressure Protection System (LTOPS) setpoint allowable values, and the LTOPS Tenable values. In addition, the cumulative core burnup applicability limits for the LTOPS would be extended.

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

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

The proposed changes modify the North Anna Units 1 and 2 RCS P/T limit curves, LTOPS setpoint allowable values, LTOPS Tenable and extend the cumulative core burnup applicability limits for the LTOPS. The allowable operating pressures and temperatures under the proposed RCS P/T limit curves are not significantly different from those allowed under the existing Technical Specification P/T limits. The revisions in the values for the LTOPS setpoint allowable values and LTOPS Tenable values do not significantly change the plant operating space. No changes to plant systems, structures or components are proposed, and no new operating modes are established. The P/T limits, LTOPS setpoint allowable values, and Tenable values do not contribute to the probability of occurrence or consequences of accidents previously analyzed. The revised licensing basis analyses utilize acceptable analytical methods, and continue to demonstrate that established accident analysis acceptance criteria are met. Therefore, there is no increase in the probability or consequences of any accident previously evaluated.

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

The proposed changes modify the North Anna Units 1 and 2 RCS P/T limit curves, LTOPS setpoint allowable values, LTOPS Tenable values and extend the cumulative core burnup applicability limits for the LTOPS. The allowable operating pressures and temperatures under the proposed RCS P/T limit curves are not significantly different from those allowed under the existing Technical Specification P/T limits. No changes to plant systems, structures or components are proposed, and no new operating modes are established. Therefore, the proposed changes do not create the possibility of any accident or malfunction of a different type previously evaluated.

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

The proposed revised RCS P/T limit curves, LTOPS setpoint allowable values, and LTOPS Tenable analysis bases do not involve a significant reduction in the margin of safety for these parameters. The effects of RCS pressure and temperature measurement uncertainty continue to be considered in the supporting analyses. The proposed revised RCS P/T limit curves are valid to cumulative core burnups of 50.3 EFPY [effective full-power year] and 52.3 EFPY for North Anna Units 1 and 2 respectively. The proposed revised LTOPS setpoint allowable values and Tenable analyses support these same cumulative core burnup limits. The analyses demonstrate that established analysis acceptance criteria continue to be met. Specifically, the proposed P/T limit curves, LTOPS setpoint allowable values and LTOPS Tenable values provide acceptable margin to vessel fracture under both normal operation and LTOPS design basis (mass addition and heat addition) accident conditions. Therefore, the proposed changes do not result in a significant reduction in 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: Ms. Lillian M. Cuoco, Esq., Senior Counsel, Dominion Resources Services, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385.

NRC Section Chief: Mary Jane Ross-Lee (Acting).

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: July 22, 2004.

Description of amendment request: The proposed change would revise Technical Specification (TS) Figure 3.5.5–1, ''Seal Injection Flow Limits,'' to reflect flow limits that allow a higher seal injection flow for a given differential pressure between the charging discharge header and the reactor coolant system pressure. Specifically, the licensee requests approval of the proposed amendment to allow for repositioning the seal injection throttle valves during the upcoming refueling outage.

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

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

The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of Emergency Core Cooling System (ECCS) flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during safety injection. The intent of the Limiting Condition for Operation (LCO) limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure sufficient centrifugal charging pump injection flow is directed to the Reactor Coolant System (RCS) via the injection points.

There are no hardware changes nor are there any changes in the method by which any safety related plant system performs its safety function. The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which [the] plant is operated and maintained. The proposed change does not alter or prevent the ability of structures, systems, and components from performing their intended safety function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed change does not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed change is consistent with the safety analysis assumptions and resultant consequences.

Since the change continues to ensure 100 percent of the assumed charging flow is available, the proposed change does not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

There are no hardware changes nor are there any changes in the method by which any safety related plant system performs its safety function. This amendment will not affect the normal method of plant operation. The proposed change does not introduce any new equipment into the plant or alter the manner in which existing equipment will be operated. No performance requirements or response time limits will be affected. The change is consistent with assumptions made in the safety analysis and licensing basis regarding limits on RCP seal injection flow.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this amendment. The[re] will be no adverse effect or challenges imposed on any safety related system as a result of this amendment.

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

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

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection function. Increasing the total seal injection flow limit to 90 gpm does not significantly impact the assumed ECCS flow that would be available for injection into the RCS following an accident.

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: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Stephen Dembek.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: July 23, 2004.

Description of amendment request: The proposed amendment will delete the requirements from the technical specifications (TS) to maintain hydrogen recombiners and hydrogen monitors. Licensees were generally required to implement upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide (RG) 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI Unit 2. Requirements related to combustible gas control were imposed by Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. The revised 10 CFR 50.44, ''Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors,'' eliminated the requirements for hydrogen recombiners and relaxed safety classifications and licensee commitments to certain design and qualification criteria for hydrogen and oxygen monitors.

The proposed license amendment will revise TS 3.3.3, ''Post Accident Monitoring (PAM) Instrumentation,'' to delete the Note in Condition C. Also in TS 3.3.3, Condition D will be deleted. In TS Table 3.3.3–1, Function 10, ''Containment Hydrogen Concentration Level,'' is deleted and replaced with ''Not Used.'' TS 3.6.8, ''Hydrogen Recombiners,'' will be deleted and the Table of Contents will be revised to reflect that deletion. TS 5.6.8, ''PAM Report,'' will be revised to reflect changing Condition G to Condition F in TS 3.3.3.

The NRC staff issued a notice of availability of a model no significant hazards consideration determination for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated July 23, 2004.

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

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

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

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

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

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

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

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

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

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

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

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

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

Therefore, this change does not involve a significant reduction in the margin of safety. Removal of hydrogen monitoring from TS will not result in a significant reduction in their functionality, reliability, and availability.

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

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Stephen Dembek.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: July 23, 2004.

Description of amendment request: The requested change will delete Technical Specification (TS) 5.6.1, ''Occupational Radiation Exposure Report,'' and TS 5.6.4, ''Monthly Operating Reports.'' The Table of

Contents will also be revised to reflect the deletions.

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on June 23, 2004 (69 FR 35067). The licensee affirmed the applicability of the model NSHC determination in its application dated July 23, 2004.

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

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

Response: No.

The proposed change eliminates the Technical Specifications (TSs) reporting requirements to provide a monthly operating letter report of shutdown experience and operating statistics if the equivalent data is submitted using an industry electronic database. It also eliminates the TS reporting requirement for an annual occupational radiation exposure report, which provides information beyond that specified in NRC regulations. The proposed change involves no changes to plant systems or accident analyses. As such, the change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accidents or transients. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. 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.

This is an administrative change to reporting requirements of plant operating information and occupational radiation exposure data, and has no effect on plant equipment, operating practices or safety analyses assumptions. For these reasons, the proposed change does not involve a significant reduction in the margin of safety.

Based upon the reasoning presented above, the requested change does not involve significant hazards consideration.

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Stephen Dembek.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: March 23, 2004.

Brief description of amendment: The amendment eliminates the Technical Specification requirements related to hydrogen monitors.

Date of Issuance: August 9, 2004.

Effective date: August 9, 2004 and shall be implemented within 60 days of issuance.

Amendment No.: 246.

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

*Date of initial notice in **Federal Register:*** April 27, 2004 (69 FR 22879).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated August 9, 2004.

No significant hazards consideration comments received: No.

Duke Energy Corporation, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

DukeEnergy Corporation, Docket Nos.50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: March 23, 2004.

Brief description of amendments: The amendments revise the reactor coolant pump flywheel inspection interval from 10 years to 20 years.

Date of issuance: August 5, 2004.

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

Amendment Nos.: 216 and 210, 223 and 205.

Renewed facility operating license Nos. NPF–35, NPF–52, NPF–9, And NPF–17: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 25, 2004.

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

No significant hazards consideration comments received: No.

Duke Energy Corporation, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of application of amendments: August 22, 2002, as supplemented by letters dated September 12, 2003, and February 4, February 16, March 23, April 28, June 17, July 6, July 12, July 19, and July 29, 2004.

Brief description of amendments: The amendments revised Technical Specification 3.8.1, ''AC Sources—Operating,'' to temporarily extend the Completion Times (CTs) for the Keowee hydro units (KHUs) to allow additional time for maintenance and upgrades. The amendments extend by 17 days (from 45 days to 62 days) the CT when one KHU is not operable and extend by 120 hours (from 60 hours to 180 hours) the CT when both KHUs are not operable.

Date of Issuance: August 5, 2004.

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

Amendment Nos.: 339, 341, and 340.

Renewed Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** September 17, 2002 (67 FR 58641).

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

No significant hazards consideration comments received: No.

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

Date of amendment request: October 21, 2003.

Brief description of amendment: The change removes MODE restrictions that prevent performance of Surveillance Requirements (SRs) 3.8.4.7 and 3.8.4.8 for the Division III direct current electrical power subsystem while in MODES 1, 2, or 3. These surveillances verify that the battery capacity is adequate to perform its required functions. The changes allow the performance of SR 3.8.4.7 and SR 3.8.4.8 during normal plant operations rather than only during refueling outages.

Date of issuance: August 12, 2004.

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

Amendment No.: 141.

Facility Operating License No. NPF–47: The amendment revised the Technical Specfications.

Date of initial notice in ***Federal Register:*** December 9, 2003 (68 FR 68662).

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

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: March 9, 2004.

Brief description of amendment: The amendment extends the completion time (CT) from 1 hour to 24 hours for Condition B of Technical Specification (TS) 3.5.1, which defines requirements for the emergency core cooling system accumulators. Condition B of TS 3.5.1 specifies a CT to restore an accumulator to operable status when it has been declared inoperable for a reason other than the boron concentration of the water in the accumulator not being within the required range.

Date of issuance: August 18, 2004.

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

Amendment No.: 222.

Facility Operating License No. DPR–64: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 13, 2004 (69 FR 19567).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 18, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 16, 2002, as supplemented March 25, 2003, April 6, and July 22, 2004.

Brief description of amendment: This amendment deleted the existing requirements in Technical Specification (TS) 3.10.D.1.d from TS 3/4.10.D, ''Multiple Control Rod Removal,'' and the associated Surveillance Requirement 4.10.D.1.d. This amendment added a new requirement to TS 3.10.D.1.d. Additionally, this amendment made an editorial change to correct a reference to TS 3.3.B.3 instead of TS 3.3.B.4 in TS 3/4.10.D.1.

Date of issuance: August 17, 2004.

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

Amendment No.: 207.

Facility Operating License No. DPR–35: Amendment revised the TSs.

Date of initial notice in ***Federal Register:*** December 10, 2002 (67 FR 75873).

The supplements dated March 25, 2003, April 6, and July 22, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 17, 2004.

No significant hazards consideration comments received: No.

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

Date of amendment request: February 9, 2004.

Brief description of amendment: The amendment eliminates the requirements in the Technical Specifications associated with hydrogen recombiners and hydrogen monitors.

Date of issuance: August 12, 2004.

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

Amendment No.: 222.

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

Date of initial notice in ***Federal Register:*** March 30, 2004 (69 FR 16617).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 9, 2004.

Brief description of amendment: The amendment eliminates the requirements in the Technical Specifications associated with hydrogen recombiners and hydrogen monitors.

Date of issuance: August 5, 2004.

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

Amendment No.: 254.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 30, 2004 (69 FR 16618).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 27, 2003.

Brief description of amendments: The amendments change Technical Specification 4.0.3, ''Missed Surveillance Time Allowance.'' TS 4.0.3 describes the relationship between meeting the surveillance requirement and operability. The amendments modify TS 4.0.3 to allow a missed surveillance to be completed within 24 hours or up to the limit of the specified interval, whichever is greater. Additionally, the amendments add a statement that a risk evaluation shall be performed for any surveillance delayed greater than 24 hours and that the risk impact shall be managed. The amendments also change the Bases to further clarify the provisions of the TS. In addition, the proposed amendments make format changes to improve appearance. The changes to the TS and its Bases are consistent with industry/Technical Specification Task Force TSTF–358, Revision 6, which was approved by the Nuclear Regulatory Commission (NRC) on October 3, 2001, and incorporated the NRC's comments on TSTF–358, Revision 5. TSTF–358, Revision 5, was approved with comment by the NRC as a part of the Consolidated Line Item Improvement Process in a **Federal Register** Notice dated September 28, 2001.

Date of issuance: August 9, 2004.

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

Amendment Nos.: 282, 266.

Facility Operating License Nos. DPR–58 and DPR–74: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 11, 2004 (69 FR 26190).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 9, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 30, 2004.

Brief description of amendments: The amendments relocate the requirements for hydrogen monitors to the Technical Requirements Manual.

Date of issuance: August 13, 2004.

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

Amendment Nos.: 214 and 219.

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

*Date of initial notice in **Federal Register:*** March 2, 2004 (69 FR 9862).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 13, 2004.

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: March 25, 2004, as supplemented June 2, 2004.

Brief description of amendments: The amendments approve a change to the licensing basis to allow the use of the methods described in Framatome–ANP Topical Report BAW–10169–A, ''RSG Plant Safety Analysis—B&W Safety Analysis Methodology for Recirculating Steam Generator Plants,'' dated October 1989, for calculating the mass and energy release rates resulting from a postulated main steamline break accident for input to containment analyses. These methods utilize the RELAP5/MOD2–B&W code approved by the Nuclear Regulatory Commission staff in a safety evaluation report dated March 14, 1995.

Date of issuance: August 19, 2004.

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

Amendment Nos.: 164 and 155.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments authorized revision to the Updated Final Safety Analysis Report.

*Date of initial notice in **Federal Register:*** April 27, 2004 (69 FR 22881).

The June 2, 2004, supplemental letter contained clarifying information and did not change the initial proposed no significant hazards consideration determination and was within the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a safety evaluation dated August 19, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 23, 2003.

Brief description of amendment: Revised the near end-of-life Moderator Temperature Coefficient (MTC) Surveillance Requirement 4.1.1.3.b by placing a set of conditions on core operation, which if met, would allow exemption from the required MTC measurement. The conditional exemption is determined on a cycle-specific basis by considering the margin predicted to the surveillance requirement MTC limit and the performance of other core parameters, such as beginning of life MTC measurements and the critical boron concentration as a function of cycle life.

Date of issuance: July 21, 2004.

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

Amendment No.: 169.

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

*Date of initial notice in **Federal Register:*** September 30, 2003 (68 FR 56346).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 21, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 30, 2003.

Brief description of amendments: The amendments revised the staff position titles in Section 5.0 ''Administrative Controls'' of the Technical Specifications.

Date of issuance: June 3, 2004.

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

Amendment Nos.: 242 and 185.

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

*Date of initial notice in **Federal Register:*** March 2, 2004 (69 FR 9865).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 3, 2004.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 23rd day of August 2004.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. 04–19586 Filed 8–30–04; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–50241; File No. SR–Amex–2004–57]

Self-Regulatory Organizations; Notice of Filing and Order Granting Accelerated Approval of a Proposed Rule Change by the American Stock Exchange LLC Relating to the Listing and Trading of Notes Linked to the Performance of the Standard & Poor's 500 Stock Index

August 24, 2004.

Pursuant to Section 19(b)(1) of the Securities Exchange Act of 1934 (''Act''),[1] and Rule 19b–4 thereunder,[2] notice is hereby given that on July 27, 2004, the American Stock Exchange LLC (''Amex'' or ''Exchange'') filed with the Securities and Exchange Commission (''SEC'' or ''Commission'') the proposed rule change as described in Items I and II below, which Items have been prepared by the Exchange. The Commission is publishing this notice to solicit comments on the proposed rule change from interested persons and is approving the proposal on an accelerated basis.

I. Self-Regulatory Organization's Statement of the Terms of Substance of the Proposed Rule Change

The Exchange proposed to list and trade notes, the performance of which is linked to the Standard & Poor's 500 Index (''S&P 500'' or ''Index'').

II. Self-Regulatory Organization's Statement of the Purpose of, and Statutory Basis for, the Proposed Rule Change

In its filing with the Commission, the Amex included statements concerning the purpose of, and basis for, the proposed rule change. The text of these statements may be examined at the places specified in Item III below. The Amex has prepared summaries, set forth in Sections A, B, and C below, of the most significant aspects of such statements.

A. Self-Regulatory Organization's Statement of the Purpose of, and the Statutory Basis for, the Proposed Rule Change

1. Purpose

Under Section 107A of the Amex Company Guide (''Company Guide''), the Exchange may approve for listing and trading securities which cannot be readily categorized under the listing criteria for common and preferred stocks, bonds, debentures, or warrants.[3] The Amex proposes to list for trading under Section 107A of the Company Guide notes linked to the performance of the S&P 500 (the ''S&P Notes'' or ''Notes'').[4] Wachovia will issue the Notes under the name ''LUNARS,'' ''Leveraged Upside Indexed Accelerated Return Securities.'' Each Note will be offered at an original public offering price of $1,000. The S&P 500 is determined, calculated and maintained solely by S&P.[5] At maturity the Notes will provide for a multiplier of any positive performance of the S&P 500 during such term subject to a maximum payment amount or ceiling to be determined at the time of issuance (the ''Capped Amount''). The Capped Amount is expected to be $1,125.[6]

The S&P 500 Notes will conform to the initial listing guidelines under Section 107A[7] and continued listing guidelines under Sections 1001–1003[8] of the Company Guide. The Notes are senior non-convertible debt securities of Wachovia. The Notes will have a term of not less than one or more than ten years. Wachovia will issue the Notes in denominations of whole units (a ''Unit'') with each Unit representing a single Note. The original public offering price will be $1,000 per Unit. The Notes will entitle the owner at maturity to receive

[1] 15 U.S.C. 78s(b)(l).

[2] 17 CFR 240. 19b–4.

[3] *See* Securities Exchange Act Release No. 27753 (March 1, 1990), 55 FR 8626 (March 8, 1990) (order approving File No. SR–Amex–89–29).

[4] Wachovia Corporation (''Wachovia'') and Standard & Poor's Corporation, a division of the McGraw-Hill Companies, Inc. (''S&P'') have entered into a non-exclusive license agreement providing for the use of the S&P 500 by Wachovia and certain affiliates and subsidiaries in connection with certain securities including these Notes. S&P is not responsible and will not participate in the issuance and creation of the Notes.

[5] The S&P 500 Index is a broad-based stock index, which provides an indication of the performance of the U.S. equity market. The Index is a capitalization-weighted index reflecting the total market value of 500 widely held component stocks relative to a particular base period. The Index is computed by dividing the total market value of the 500 stocks by an Index divisor. The Index Divisor keeps the Index comparable over time to its base period of 1941–1943 and is the reference point for all maintenance adjustments. The securities included in the Index are listed on the Amex, New York Stock Exchange, Inc. (''NYSE'') or traded through NASDAQ. The Index reflects the price of the common stocks of 500 companies without taking into account the value of the dividend paid on such stocks. The Index Value is disseminated once every fifteen seconds through numerous data providers. Telephone conference between Laura Clare, Assistant General Counsel, Amex, and Florence Harmon, Senior Special Counsel, Division of Market Regulation (''Division''), Commission, on August 20, 2004 (pertaining to dissemination of Index Value).

In connection with the S&P 500, the Exchange notes that S&P has announced a change to its methodology so that weightings are based on the ''public float'' of a component stock and not those shares of stock that are not publicly traded. The S&P 500 is currently a market capitalization weighted index that is expected to be changed to a ''float-adjusted'' market capitalization index by September 2005. In a ''traditional'' market capitalization index, the value of the index is calculated by multiplying the total number of shares outstanding of each component by the price per share of the component. The result is then divided by the divisor. On March 1, 2004, S&P announced that it intends to shift its major indexes, such as the S&P 500 to a ''float-adjusted'' market capitalization index. In a ''float-adjusted'' market capitalization index, the value of the index will be calculated by multiplying the public float of each component by the price per share of the component. The result is then divided by the divisor. Accordingly, a ''float-adjusted'' market capitalization index will exclude those blocks of stocks that do not publicly trade from determining the weight for a stock in the index. The transition from a market capitalization weighted index to a ''float-adjusted'' capitalization weighted index will be implemented over an 18-month period.

[6] *See* prospectus supplement dated August 3, 2004.

[7] The initial listing standards for the Notes require: (1) A minimum public distribution of one million units; (2) a minimum of 400 shareholders; (3) a market value of at least $4 million; and (4) a term of at least one year. In addition, the listing guidelines provide that the issuer has assets in excess of $100 million, stockholder's equity of at least $10 million, and pre-tax income of at least $750,000 in the last fiscal year or in two of the three prior fiscal years. In the case of an issuer that is unable to satisfy the earning criteria stated in Section 101 of the Company Guide, the Exchange will require the issuer to have the following: (1) Assets in excess of $200 million and stockholders' equity of at least $10 million; or (2) assets in excess of $100 million and stockholders' equity of at least $20 million. The Exchange concluded, pursuant to its evaluation of the nature and complexity of the product pursuant to Section 107A, not to issue a circular regarding member firm compliance responsibilities because the notes are issued in $1,000 denominations and are categorized as debt. Telephone conference between Jeffrey Burns, Associate General Counsel, Amex, and Florence Harmon, Senior Special Counsel, Division, Commission, on August 24, 2004 (pertaining to issuance of a circular to members).

[8] The Exchange's continued listing guidelines are set forth in Sections 1001 through 1003 of Part 10 to the Exchange's Company Guide. Section 1002(b) of the Company Guide states that the Exchange will consider removing from listing any security where, in the opinion of the Exchange, it appears that the extent of public distribution or aggregate market value has become so reduced to make further dealings on the Exchange inadvisable. With respect to continued listing guidelines for distribution of the Notes, the Exchange will rely, in part, on the guidelines for bonds in Section 1003(b)(iv) because the Notes are issued in $1,000 denominations. Section 1003(b)(iv)(A) provides that the Exchange will normally consider suspending dealings in, or removing from the list, a security if the aggregate market value or the principal amount of bonds publicly held is less than $400,000.