NUCLEAR REGULATORY COMMISSION

[Docket No. 50–346]

FirstEnergy Nuclear Operating Company, Davis-Besse Nuclear Power Station, Unit 1, Withdrawal of Exemption

1.0 Background

The FirstEnergy Nuclear Operating Company (the licensee) is the holder of Facility Operating License No. NPF–3 which authorizes operation of the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS). The license provides, among other things, that the facility is subject to all rules, regulations, and orders of the Nuclear Regulatory Commission (NRC, the Commission) now or hereafter in effect.

The facility consists of a pressurized-water reactor located in Ottawa County in Ohio.

2.0 Request

Title 10 of the Code of Federal Regulations (10 CFR), part 50, appendix R, subsection III.L.1 requires that alternative or dedicated shutdown capability be able to achieve cold shutdown conditions within 72 hours. The NRC granted an exemption to this requirement by letter dated August 20, 1984, for DBNPS.

In summary, the licensee now concludes that DBNPS meets the requirement and the exemption is no longer required; therefore, the licensee requests that the exemption be withdrawn.

3.0 Evaluation

Two issues caused the licensee to originally request the exemption. They were the ability to depressurize the reactor coolant system and a limitation on cooldown rate. The licensee has recently performed an evaluation and determined that alternate pressurizer spray from the high pressure injection pumps could be used for depressurization and the limit on cooldown rate can be increased. The licensee concluded that DBNPS can now comply with the regulation and the exemption is no longer required.

Based upon the licensee's recent evaluation determining that DBNPS alternative shutdown capability can achieve cold shutdown within 72 hours, the staff concludes that the exemption can be withdrawn.

4.0 Conclusion

Accordingly, the Commission hereby grants FirstEnergy Nuclear Operating Company withdrawal of the exemption from the requirements of CFR part 50, appendix R, subsection III.L.1, granted by letter dated August 20, 1984, for DBNPS.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption withdrawal will not have a significant effect on the quality of the human environment (69 FR 28951).

This exemption withdrawal is effective upon issuance.

Dated at Rockville, Maryland, this 24th day of June, 2004.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

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NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued, from June 11, 2004, through June 23, 2004. The last biweekly notice was published on June 22, 2004 (69 FR 34696).

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 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

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

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

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

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

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

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

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

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

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

Date of amendment request: May 20, 2004.

Description of amendment request: The proposed amendment would support conversion from an 18-month to a 24-month fuel cycle. Specifically, the proposed amendment would (1) change certain technical specification (TS) surveillance requirement (SR) frequencies from ''18 months'' to ''24 months'' in accordance with the guidance of Generic Letter 91–04, ''Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle,'' (2) change Administrative Controls Section 5.5.7, ''Ventilation Filter Testing Program (VFTP),'' to address changes to 18-month frequencies that are specified in Regulatory Guide 1.52, ''Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,'' and (3) change selected allowable values for instrumentation setpoints. In addition, two separate administrative changes are being proposed to eliminate temporary changes that have expired and no longer apply. These include (1) removal of TS Table 3.0.2–1, ''Surveillance Intervals Extended to November 30, 2000,'' and a reference to it in SR 3.0.2, and (2) removal of footnotes (a) and (b) from TS Table 3.3.8.1–1, ''Loss of Power 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 amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed TS changes involve a change in the surveillance testing intervals and allowable values to facilitate a change in the operating cycle length. The analytical limit increase for the Reactor Vessel Pressure-High function remains conservative with respect to considerations for isolating the Residual Heat Removal-Shut Down Cooling (RHR–SDC) system in the event of a line break and for providing overpressure protection to the low pressure RHR–SDC system piping. Also included in this application are administrative changes to remove Table 3.0.2–1 and the reference to it in SR 3.0.2 (since this implements an expired one-time TS exception), to renumber certain SRs remaining at 18 month frequencies, and to remove footnotes (a) and (b) from Table 3.3.8.1–1 that applied temporary allowable values until completion of modification to tap settings and degraded voltage setpoints. The proposed TS changes do not physically impact the plant. The proposed TS changes do not degrade the performance of, or increase the challenges to, any safety systems assumed to function in the accident analysis. The proposed TS changes do not impact the usefulness of the SRs in evaluating the operability of required systems and components, or the way in which the surveillances are performed. In addition, the frequency of surveillance testing is not considered an initiator of any analyzed accident, nor does a revision to the frequency introduce any accident initiators. The specific value of the allowable value is not considered an initiator of any analyzed accident. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The consequences of a previously evaluated accident are not significantly increased. The proposed change does not affect the performance of any equipment credited to mitigate the radiological consequences of an accident. Evaluation of the proposed TS changes demonstrated that the availability of credited equipment is 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 equipment. Historical review of surveillance test results and associated maintenance records did not find evidence of failures that would invalidate the above conclusions.

The allowable values have been developed in accordance with RG 1.105, ''Instrument Setpoints,'' to ensure that the design and safety analysis limits are satisfied. The methodology used for the development of the allowable values ensures the affected instrumentation remains capable of mitigating design basis events as described in the safety analyses and that the results and radiological consequences described in the safety analyses remain bounding. Therefore, the proposed change does not alter the ability to detect and mitigate events and, as such, does not involve a significant increase in the consequences of an accident previously evaluated.

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

Response: No.

The proposed TS changes involve a change in the surveillance testing intervals and allowable values to facilitate a change in the operating cycle length. The analytical limit increase for the Reactor Vessel Pressure-High function remains conservative with respect to considerations for isolating the RHR–SDC system in the event of a line break and for providing overpressure protection to the low pressure RHR–SDC system piping. Also included in this application are administrative changes to remove Table 3.0.2–1 and the reference to it in SR 3.0.2 since this implements an expired one-time exception, to renumber certain SRs remaining at 18 month frequencies, and to remove footnotes (a) and (b) from Table 3.3.8.1–1 that applied temporary allowable values until completion of modification to tap settings and degraded voltage setpoints. The proposed TS changes do not introduce any failure mechanisms of a different type than those previously evaluated, since there are no physical changes being made to the facility. No new or different equipment is being installed. No installed equipment is being operated in a different manner. As a result, no new failure modes are being introduced. The way surveillance tests are performed remains unchanged. A historical review of surveillance test results and associated maintenance records indicated there was no evidence of any failures that would invalidate the above conclusions.

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

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

Response: No.

The proposed TS changes involve a change in the surveillance testing intervals and allowable values to facilitate a change in the operating cycle length. The analytical limit increase for the Reactor Vessel Pressure-High function remains conservative with respect to considerations for isolating the RHR–SDC system in the event of a line break and for providing overpressure protection to the low pressure RHR–SDC system piping. Also included in this application are administrative changes to remove Table 3.0.2–1 and the reference to it in SR 3.0.2 since this implements an expired one-time exception, to renumber certain SRs remaining at 18 month frequencies, and to remove footnotes (a) and (b) from Table 3.3.8.1–1 that applied temporary allowable values until completion of modification to tap settings and degraded voltage setpoints. The impact of these changes on system availability is not significant, based on other more frequent testing that is performed, the existence of redundant systems and equipment, and overall system reliability. Evaluations have shown there is no evidence of time dependent failures that would impact the availability of the systems. The proposed changes do not significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation. The proposed changes in TS instrumentation allowable values are the result of application of the CPS setpoint methodology using plant specific drift values. The revised allowable values more accurately reflect total instrumentation loop accuracy including drift while continuing to protect any assumed analytical limit. The proposed changes do not result in any hardware changes or in any changes to the analytical limits assumed in accident analyses. Existing operating margin between plant conditions and actual plant setpoints is not significantly reduced due to these changes. The proposed changes do not significantly impact any safety analysis assumptions or results.

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

Date of amendment request: May 20, 2004.

Description of amendment request: The licensee proposed to revise the Technical Specifications (TS), section 3.2.B.4, to clarify the application of the action requirements for inoperable control rods. Specifically, this involves adding wording to clarify that operable control rods that have been taken out of service at the fully inserted position (*i.e.*, disarmed) to perform hydraulic control unit maintenance are not to be counted as inoperable control rods. Control rods that have been fully inserted, and disarmed, fulfill the safety function of the control rod since it is in a position of maximum contribution to shutdown reactivity. Such clarification is consistent with the intent of the current operability requirements, and with the Nuclear Regulatory Commission guidance document entitled ''Standard Technical Specifications—General Electric Plants, BWR [Boiling Water Reactor]/4,'' NUREG–1433, Revision 2, where the control rod operability requirements explicitly apply to ''inoperable control rods'' and ''withdrawn stuck control rods.''

In addition, the licensee proposed to correct a typographical error in Table 3.1.1 (page 3.1–12 of the TS), where ''note i'' was inadvertently typed as ''note I.''

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 proposed amendment involves clarifying, but not changing, the current intent of control rod operability requirements. The proposed amendment also corrects a typographical error. These changes will not lead to alteration of the physical design or operational procedures associated with the control rod system, or any other plant structure, system, or component (SSC). All requirements needed to assure the operability of the control rod system will remain unchanged. Action requirements for control rods were not assumed to be precursors of accidents, nor were they assumed to be components 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 changes involve clarification of control rod operability requirements and correction of a typographical error. 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: Thomas S. O'Neill, Associate General Counsel, Exelon Generation Company, LCC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Section Chief: Richard J. Laufer.

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

Date of amendments request: February 4, 2004.

Description of amendments request: The amendments would revise Technical Specification (TS) 3.7.1, ''Main Steam Safety Valves (MSSVs),'' to: (1) Permit operation in Mode 3 with 5 to 8 inoperable MSSVs (2 to 5 operable MSSVs) per steam generator, (2) increase the completion time to reduce the variable overpower trip (VOPT) setpoint when 1 to 4 MSSVs per steam generator are inoperable, and (3) make associated editorial changes.

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

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

Response: No. Each change is discussed below.

• Revise Technical Specification (TS) 3.7.1 to permit operation in Mode 3 when there are five to eight inoperable MSSVs (two to five operable MSSVs) per steam generator.

This proposed change would allow the plant to remain in Mode 3 with as few as two operable MSSVs per steam generator. Currently, the plant must be placed in Mode 4 with fewer than six operable MSSVs per steam generator. Two MSSVs have sufficient relieving capacity to dissipate core decay heat and reactor coolant pump heat in Mode 3 to limit secondary system pressure to less than or equal to 110% of design pressure, as required by ASME Code, Section III. A minimum of two MSSVs per steam generator (four total) would be required to be operable in Mode 3 in case of a single failure of one of the valves. Since this proposed change would continue to provide over-pressure protection and heat removal capability in Mode 3, this change would have no affect on any analyzed accidents. Therefore, this proposed change would not involve a significant increase in the probability or consequences of an accident previously evaluated.

• Increase the Completion Time for Required Action A.2 of TS 3.7.1 (reduce the variable overpower trip [VOPT] setpoint when one to four MSSVs per steam generator are inoperable) from 12 hours to 36 hours.

Required Action A.2 of TS 3.7.1 specifies a Completion Time of 12 hours to reduce the variable overpower trip (VOPT)—high setpoint if one or more required MSSVs are inoperable. The proposed increase in the Completion Time for Action A.2 from 12 hours to 36 hours is consistent with Industry/Technical Specification Task Force TSTF–235, Revision 1, incorporated in Revision 2 of NUREG–1432, Combustion Engineering Standard Technical Specifications. The revised TS 3.7.1 Bases associated with TSTF–235, Revision 1, states that the Completion Time of 36 hours for Required Action A.2 is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period. Increasing the Completion Time to reset the VOPT from

12 hours to 36 hours does not involve a significant increase in the probability or consequences of an accident previously evaluated.

• Make associated editorial changes.

The associated editorial changes do not change any structure, system or component (SSC) or affect the operation or maintenance of any SSC. They are editorial enhancements to make the TSs easier to understand, eliminate potential inconsistencies with other TSs, and reduce the potential for human errors. 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.

Response: No. Each change is discussed below.

• Revise Technical Specification (TS) 3.7.1 to permit operation in Mode 3 when there are five to eight inoperable MSSVs (two to five operable MSSVs) per steam generator.

This proposed change would allow the plant to remain in Mode 3 with as few as two operable MSSVs per steam generator. Currently, the plant must be placed in Mode 4 with fewer than six operable MSSVs per steam generator. Two MSSVs have sufficient relieving capacity to dissipate core decay heat and reactor coolant pump heat in Mode 3 to limit secondary system pressure to less than or equal to 110% of design pressure, as required by ASME Code, Section III. A minimum of two MSSVs per steam generator (four total) would be required to be operable in Mode 3 in case of a single failure of one of the valves. This proposed change would continue to provide overpressure protection and heat removal capability in Mode 3. Therefore, this proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

• Increase the Completion Time for Required Action A.2 of TS 3.7.1 (reduce the variable overpower trip [VOPT] setpoint when one to four MSSVs per steam generator are inoperable) from 12 hours to 36 hours.

Required Action A.2 of TS 3.7.1 specifies a Completion Time of 12 hours to reduce the variable overpower trip—high setpoint if one or more required MSSVs are inoperable. The proposed increase in the Completion Time for Action A.2 from 12 hours to 36 hours is consistent with Industry/Technical Specification Task Force TSTF–235, Revision 1, incorporated in Revision 2 of NUREG–1432, Combustion Engineering Standard Technical Specifications. The revised TS 3.7.1 Bases associated with TSTF–235, Revision 1, states that the Completion Time of 36 hours for Required Action A.2 is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period. Therefore, this proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

• Make associated editorial changes.

The associated editorial changes do not change any structure, system or component (SSC) or affect the operation or maintenance of any SSC. They are editorial enhancements to make the TSs easier to understand, eliminate potential inconsistencies with other TSs, and reduce the potential for human errors. 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.

Response: No. Each change is discussed below.

• Revise Technical Specification (TS) 3.7.1 permit operation in Mode 3 when there are five to eight inoperable MSSVs (two to five operable MSSVs) per steam generator.

This proposed change would allow the plant to remain in Mode 3 when there are as few as two operable MSSVs per steam generator. Currently, the plant must be placed in Mode 4 with fewer than six operable MSSVs per steam generator. Two MSSVs have sufficient relieving capacity to dissipate core decay heat and reactor coolant pump heat in Mode 3 to limit secondary system pressure to less than or equal to 110% of design pressure, as required by ASME Code, Section III. A minimum of two MSSVs per steam generator (four total) would be required to be operable in Mode 3 in case of a single failure of one of the valves. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

• Increase the Completion Time for Required Action A.2 of TS 3.7.1 (reduce the variable overpower trip [VOPT] setpoint when one to four MSSVs per steam generator are inoperable) from 12 hours to 36 hours.

Required Action A.2 of TS 3.7.1 specifies a Completion Time of 12 hours to reduce the variable overpower trip—high setpoint if one or more required MSSVs are inoperable. The proposed increase in the Completion Time for Action A.2 from 12 hours to 36 hours is consistent with Industry/Technical Specification Task Force TSTF–235, Revision 1, incorporated in Revision 2 of NUREG–1432, Combustion Engineering Standard Technical Specifications. The revised TS 3.7.1 Bases associated with TSTF–235, Revision 1, states that the Completion Time of 36 hours for Required Action A.2 is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

• Make associated editorial changes.

The associated editorial changes do not change any structure, system or component (SSC) or affect the operation or maintenance of any SSC. They are editorial enhancements to make the TSs easier to understand, eliminate potential inconsistencies with other TSs, and reduce the potential for human errors. 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 that review, it appears that the three 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: Kenneth C. Manne, Senior Attorney, Arizona Public Service Company, P.O. Box 52034, Mail Station 7636, Phoenix, Arizona 85072–2034.

NRC Section Chief: Stephen Dembek.

Dominion Nuclear Connecticut Inc., et al., Docket No. 50–423, Millstone Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: April 15, 2004.

Description of amendment request: The proposed amendment would modify the fire protection license condition to reflect a proposed permanent change to the CO_2 fire suppression system in the cable spreading area (CSA).

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:

Criterion 1:

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

Response: No.

The CO_2 system is designed to limit the effects of fire damage to plant equipment and does not contribute to the prevention or initiation of a fire event. The CO_2 system is not safety-related and is not relied upon to safely shut down the reactor, mitigate radiological consequences of any accident, or maintain the reactor in a safe shutdown condition. Accordingly, the proposed amendment does not affect the inputs or assumptions for any accidents previously evaluated nor does it affect initiation of a fire event. Modifying the CO_2 initiation system to a manual mode reduces the possibility of a malfunction leading to an inadvertent CO_2 discharge. Because the automatic initiation feature of the CO_2 system would be eliminated by the proposed amendment, inadvertent operation would no longer need to be a postulated failure for the CO_2 system. The current analysis for a worst-case fire event allows for complete loss of the CSA which is protected by 3-hour fire-rated barriers. Alternate safe shutdown methods are available in the event that a fire consumes all equipment and cables in the room. The proposed amendment does not modify the fire suppression methodology in a way that would cause any greater damage than complete loss of the CSA. The incipient fire detection system offsets the delay time for manual CO_2 initiation by allowing an earlier response time by the fire brigade. Failure to take manual action is bounded by previous failure of the CO_2 system to operate. Based on this discussion, the proposed amendment does not increase the probability or consequence of an accident previously evaluated.

Criterion 2:

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

Response: No.

The CO_2 system is a mitigating system designed to limit the effects of fire damage to plant equipment and is not credited for safe shutdown of the plant. The proposed amendment does not involve any change that would impact designed CO_2 concentration levels and therefore does not affect the ability of the CO_2, once delivered, to act as [a] fire extinguishing agent. The proposed amendment does not introduce failure modes, accident initiators, or malfunctions that would cause a new or different kind of accident or fire event. The potential for increased water usage due to the proposed change in fire fighting methodology for the CSA is within the capability and capacity of the existing site fire water system and potential water buildup on the CSA floor is bounded by the existing flooding analysis. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3:

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

Response: No.

The evaluated fire event assumes a fire coincident with a loss of power, with no additional plant accidents. As stated above, the current analysis for a worst-case fire event in the CSA allows for complete loss of all cables and equipment in the CSA resulting in loss of use of the control room. The proposed amendment changes the CO_2 system initiation method from automatic to manual and impacts the response time of applying CO_2 as a fire-extinguishing agent. This impact is not significant in that any potential increase in fire damage does not exceed complete loss of all the CSA cables and equipment. In addition, the incipient fire detection system offsets the delay time for manual CO_2 initiation by allowing an earlier response time by the fire brigade. The proposed amendment does not modify the CSA fire area 3-hour fire rated barriers. Therefore, based on the above, the proposed amendment 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: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Waterford, CT 06141–5127.

NRC Section Chief: James W. Clifford.

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 amendment request: May 25, 2004.

Description of amendment request: The proposed amendments would revise the licensing basis in the Updated Final Safety Analysis Report to support installation of a passive low-pressure injection (LPI) cross connect inside containment for Unit 3. The proposed changes would revise the licensing basis for selected portions of the core flood and LPI piping to allow exclusion of the dynamic effects associated with a postulated rupture of that piping by application of leak-before-break technology. Similar amendments were approved for Unit 1 by NRC letter dated September 29, 2003, and for Unit 2 by NRC letter dated February 5, 2004.

The proposed amendments would also delete technical specifications (TSs) which will no longer apply when the LPI cross connect modification has been implemented. *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) Involve a significant increase in the probability or consequences of an accident previously evaluated:

The proposed License Amendment Request (LAR) modifies the Unit 3 licensing basis to allow the dynamic effects associated with postulated pipe rupture of selected portions of the Unit 3 Low Pressure Injection (LPI)/Core Flood (CF) piping to be excluded from the design basis. The proposed LAR also removes Technical Specifications that are no longer applicable due to the completion of the LPI cross connect modification on all three Oconee Units. The proposed design allowances for these selected portions of piping continue to allow the LPI system design to meet General Design Criteria (GDC) 4 requirements related to environmental and dynamic effects. The proposed LAR will continue to ensure that ONS [Oconee Nuclear Station] can meet design basis requirements associated with the LPI safety function. The addition of the crossover line will enhance the ability of the control room operator to mitigate the consequences of specific events for which LPI is credited. Therefore, the proposed LAR does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any kind of accident previously evaluated:

The proposed LAR modifies the Unit 3 licensing basis to allow the dynamic effects associated with postulated pipe rupture of selected portions of Unit 3 LPI/CF piping to be excluded from the design basis and removes TS requirements that are no longer applicable due to the completion of the LPI cross connect modification on all three Oconee Units. The proposed design allowances for these selected portions of piping continue to allow the LPI system design to meet GDC 4 requirements related to environmental and dynamic effects. The systems affected by the changes are used to mitigate the consequences of an accident that has already occurred. The proposed licensing basis change does not affect the mitigating function of these systems. Consequently, these changes do not alter the nature of events postulated in the Safety Analysis Report nor do they introduce any unique precursor mechanisms. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed licensing basis and TS changes do not unfavorably affect any plant safety limits, set points, or design parameters. The changes also do not unfavorably affect the fuel, fuel cladding, RCS [Reactor Coolant System], or containment integrity. Therefore, the proposed changes, which add new design allowances associated with the passive LPI cross connect modification and remove obsolete TS requirements, 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: Anne W. Cottingham, Winston and Strawn LPP, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Stephanie M. Coffin (Acting).

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: May 26, 2004.

Description of amendment request: The proposed amendments would revise the current 72-hour allowed outage time (AOT) for the emergency diesel generators (EDGs) in Technical Specification (TS) 3.8.1.1 to a 14-day AOT. Additionally, the proposed amendments delete the surveillance requirement in TS 4.8.1.1.2.b.1 which requires an EDG inspection, in accordance with the manufacturer's recommendations, every 18 months during shutdown. The periodic EDG maintenance inspection requirements will be relocated to a licensee-controlled maintenance program that is referenced in the Updated Final Safety Analysis Report (UFSAR). Future changes to the EDG maintenance program would then be controlled pursuant to Title 10 of the Code of Federal Regulations (10 CFR), Section 50.59. Lastly, the proposed amendments would revise footnote (1) of TS 3.8.1.1, which currently provides a 7-day AOT to restore EDG fuel oil properties which do not meet the requirements of TS 4.8.1.1.2.d.2 or TS 4.8.1.1.2.e. The revised footnote wording would allow delay of action requirements for up to 7 days when the EDGs are inoperable solely as a result of failure to meet TS 4.8.1.1.2.d.2 or TS 4.8.1.1.2.e surveillance requirements.

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

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

Response: No.

The proposed change does not affect the design, operational characteristics, function or the reliability of the EDGs. The EDGs are not initiating conditions for any accident previously evaluated. The EDGs mitigate the consequences of previously evaluated accidents involving loss of offsite power.

The consequences of any previously analyzed accident will not be significantly affected by extending the AOT for a single EDG, since the remaining EDG supporting the redundant Engineered Safety Features systems will continue to be available to perform the accident mitigation functions. In addition, to fully evaluate the effects of the proposed EDG AOT extension, a Probabilistic Risk Assessment was performed to quantitatively assess the risk impact of the proposed change for each unit. The results of this risk assessment concluded that the increase in plant risk is very small and consistent with the guidance contained in Regulatory Guide 1.174 and Regulatory Guide 1.177.

The deletion of TS surveillance requirement 4.8.1.1.2.b.1 from the Technical Specifications will not impact the capability of the EDGs to perform their accident mitigation functions. The required EDG maintenance inspections will continue to be performed in accordance with the licensee EDG maintenance program. The risk of performing the maintenance inspections during power operation has been considered in the EDG AOT extension supporting risk evaluation and determined to be acceptable.

The proposed change to footnote (1) of TS 3.8.1.1 will also not impact the capability of the EDGs to perform their accident mitigation functions. Fuel oil properties that are not within the specified limits will not have an immediate effect on EDG operation and restoring the fuel oil to within limits within 7 days will ensure the availability of high grade fuel oil for the EDGs.

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 changes do not involve a change in the design, configuration, or method of operation of the plant. The changes do not involve the addition of new equipment or the modification of existing equipment. As such, no new failure modes are introduced by these changes.

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

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

Response: No.

The proposed changes do not alter the plant design and do not affect any assumptions or inputs to the safety analysis. The proposed changes to the EDG allowed outage time have been evaluated both deterministically and using a risk informed approach. These evaluations demonstrate that power system design defense-in-depth capabilities will be maintained and that the risk contribution is small.

In addition, the proposed deletion of the EDG maintenance inspection surveillance requirements from the TS[s] and modifications to the EDG action requirements associated with the EDG fuel oil surveillances will not impact the EDG reliability and their capability to perform their accident mitigation function.

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, 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: June 2, 2004.

Description of amendment request: The proposed amendments would revise the Technical Specification (TS) surveillance interval from monthly to quarterly for certain reactor trip system and engineered safety feature actuation system channel functional tests in accordance with the methodology presented in the Nuclear Regulatory Commission-approved topical report, WCAP–10271, ''Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System,'' and supplements thereto.

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.

Operation of the Beaver Valley Power Station in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change modifies surveillance frequencies. Increases in the surveillance test intervals have been established based on achieving acceptable levels of equipment reliability. Consequently, equipment that is required to operate to mitigate an accident will continue to operate as expected and the probability of the initiation of any accident previously evaluated will not be significantly increased. Implementation of the proposed changes does not alter the manner in which protection is afforded. This equipment will continue to be tested in a manner and at a frequency to give confidence that the equipment can perform its assumed safety function. As a result, the proposed surveillance requirement changes do not significantly affect the consequences of any accident previously evaluated.

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 previously evaluated?

Response: No.

The proposed change does not involve any physical changes to the plant or the modes of plant operation defined in the Technical Specifications. The proposed change does not involve the addition or modification of plant equipment nor does it alter the design or operation of any plant systems. No new accident scenarios, transient precursors or failure mechanisms are introduced as a result of these changes.

There are no changes in this proposal that would cause the malfunction of safety-related equipment assumed to be operable in accident analyses. No new mode of failure has been created and no new equipment performance requirements are imposed. The proposed change has no effect on any previously evaluated accident.

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

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

Response: No.

The change in surveillance frequencies has been evaluated to ensure that it provides an acceptable level of equipment reliability. Equipment continues to be tested at a frequency that gives confidence that the equipment can perform its assumed safety function when required. The proposed changes do not alter the manner in which safety limits, limiting safety system setpoints or limiting conditions for operations are determined. The impact of reduced testing is to allow a longer time interval over which instrument uncertainties (*e.g.* drift) may act. Experience has shown that the initial uncertainty assumptions are valid for reduced testing.

Implementation of the proposed changes is expected to result in an overall improvement in safety since plant transients initiated from inadvertent safety system actuation should be reduced. Less frequent testing will reduce the likelihood for inadvertent reactor trips and inadvertent actuation of Engineered Safety Feature Actuation System components.

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: 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: March 31, 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 section 50.65(a)(4) of Title 10 of the Code of Federal Regulations (10 CFR), part 50. Limiting Condition for Operation (LCO) 3.0.4 exceptions in individual TS would be eliminated, several notes or specific exceptions are revised to reflect the related changes to LCO 3.0.4, and Surveillance Requirement 3.0.4 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 1, 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 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 E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: May 25, 2004.

Description of amendment request: The proposed amendment revises Technical Specifications (TSs) 3.10.e and 3.10.f to add an allowed outage time for the individual rod position indication (IRPI) system of 24 hours with more than one IRPI group inoperable. Additional changes add the demand step counters to the TSs and add a note to allow for a soak time subsequent to substantial rod motion for the rods that exceed their position limits before invoking the TS requirements. Also, the definition of ''immediately'' is added to TS 1.0.

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

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

Response: No.

Rod position indication instrumentation is not an assumed accident initiator, providing indication only of the control and shutdown rods position. Normal operation, abnormal occurrences and accident analyses assume the rods are at certain positions within the reactor core. The changes requested herein modify the time the existing two rod position indication systems may be inoperable and provide appropriate actions to compensate for that inoperability and add the second, digital, rod position indication system to the TS. Thus, this change does not involve a significant increase in the probability of an accident.

The condition of concern is the alignment of the rods. Operating with a rod position indicator inoperable does not change the position of the rod; an inoperable rod position indication instrument does not make a rod misaligned. An increase in the consequences with the rods only comes from a rod being misaligned such that an increase in the heat produced in a localized area causes the fuel to fail either during operation, during a plant transient or post-accident. An inoperable rod position indicator does not change the position of the rod. Rod position is subsequently verified by other means if the rod is moved by greater than a predetermined amount. Indication of rod position by other means ensures rod position remains within analytical limits. Thus, inoperable rod position indication instrumentation does not involve an increase in the consequences of an accident.

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

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

Response: No.

This proposed change does not alter the design, function, or operation of any plant component and does not install any new or different equipment. The malfunction of safety related equipment, assumed operable in the accident analyses, would not be caused because of the proposed technical specification change. No new failure mode has been created and no new equipment performance burdens are imposed. Therefore, the possibility of a new or different kind of accident from those previously analyzed has not been created.

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

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

Response: No.

The rod position indication system is an instrumentation system that provides indication to the operators that a control rod may be misaligned. Inoperable individual rod position indication instrumentation does not by itself harm or affect reactor operation, but may impair the ability of the operators to detect a misaligned rod. To compensate for this potential impairment of the operators' ability to detect a misaligned rod, requirements to verify the inoperable rod position indicators position are added. The impact of inoperable rod position indication instrumentation is offset by the availability of other indications that a rod is misaligned. Excore and incore nuclear instrumentation provides indication that reactor power, flux density, may have shifted axially or radially. Also, thermocouple indication would show that the core temperatures have increased in one region of the core and/or decreased in another region of the core.

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: 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: June 1, 2004.

Description of amendment request: The proposed amendment revises Technical Specification (TS) 1.0, ''Definitions,'' Table TS 3.5–2, ''Instrument Operation Conditions for Reactor Trip,'' and Table TS 4.1–1, ''Minimum Frequencies for Checks, Calibrations and Test of Instrument Channels.'' The TS revisions will add a definition for ''staggered test basis,'' increase surveillance test intervals for reactor protection system and engineered safety features actuation system analog channels and logic cabinets, and add a completion time for the reactor trip breakers.

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?

No. The proposed changes to the STIs [surveillance test intervals] and the RTB CT [reactor trip breaker completion time] reduce the potential for inadvertent reactor trips and spurious actuations, and therefore, do not increase the probability of an accident previously evaluated.

The proposed changes will not result in a significant increase in the risk of plant operation as demonstrated in WCAP–15376-P-A. The impact of plant safety as measured by core damage frequency (CDF) is less than 1.0E–06 per year and the impact of large early release frequency (LERF) is less than 1.0E–07 per year. For the addition of the RTB CT, the incremental conditional core damage probabilities (ICCDP) and incremental conditional large early release probabilities (ICLERP) are less than 5.0E–08. These changes meet the acceptance criteria in Regulatory Guides 1.174 and 1.177. Therefore, there will not be a significant increase in the probability of an accident.

The proposed changes did not include any hardware changes, and therefore, all structures, systems, and components will continue to perform their intended function to mitigate the consequences of an event within the assumed acceptance limits. The proposed changes do not affect source term, containment isolation, or the radiological release assumptions used in evaluating radiological consequences of previously analyzed accidents. Therefore, the proposed changes do not increase the consequences of an accident previously evaluated.

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

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

No. The proposed changes do not involve any hardware changes, any setpoint changes, any addition of safety related equipment, or any changes in the manner in which the systems provide plant protection. Additionally, all operator actions credited in accident analyses remain the same. There are no new or different accident initiators or new accidents scenarios created by the proposed changes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The safety analyses acceptance criteria in the Updated Safety Analysis Report (USAR) are not impacted by these changes. The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. All signals and operator actions credited in the USAR accident analyses will remain the same. Redundant RPS [reactor protection system] and ESFAS [engineered safety features actuation system] trains are maintained and diversity with regard to the signals that provide reactor trip and engineered safety features actuation is also maintained. The calculated impact on risk continues to meet the acceptance criteria contained in Regulatory Guides 1.174 and 1.177. Therefore, the 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: 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 Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota

Date of amendment request: May 3, 2004.

Description of amendment request: The proposed amendments would revise the Prairie Island Nuclear Generating Plant (PINGP) licensing basis to (1) define a hydraulic analysis methodology for demonstrating functionality of the cooling water (CL) system following a design basis seismic event and (2) define acceptance criteria from the American Society of Mechanical Engineers (ASME) Section III Code, Subsection ND, when performing stress analysis of the CL system non-Class I piping with design basis seismic loads.

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 license amendment proposes to revise the plant licensing basis to include: (1) a hydraulic analysis methodology for demonstrating functionality of the CL system following a design basis seismic event; and (2) American Society of Mechanical Engineers, Section III, Subsection ND, ''Class 3 Components,'' 1986 Edition, Service Level D as the basis for acceptance criteria when performing stress analysis of the cooling water system non-Class I piping with design basis seismic loads.

The cooling water system provides a heat sink for removal of process and operating heat from safety related components during design basis accidents. This system is not an accident initiator and thus these proposed licensing basis changes do not increase the probability of a previously evaluated accident.

The proposed plant licensing basis changes will provide the basis for evaluating cooling water system capability during and following a design basis seismic event. Use of the proposed methodology and acceptance criteria will conservatively demonstrate that the cooling water system will continue to provide its design cooling function. With the cooling water system design heat removal capability maintained, accident consequences will not be increased. Thus these licensing basis changes do not involve an increase in the consequences of an accident previously evaluated.

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

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

Response: No.

This license amendment proposes to revise the plant licensing basis to include: (1) a hydraulic analysis methodology for demonstrating functionality of the CL system following a design basis seismic event; and (2) American Society of Mechanical Engineers, Section III, Subsection ND, ''Class 3 Components,'' 1986 Edition, Service Level D as the basis for acceptance criteria when performing stress analysis of the cooling water system non-Class I piping with design basis seismic loads.

The proposed licensing basis changes do not involve a change in system operation, or procedures involved with the cooling water system. The proposed changes provide a conservative basis for evaluating cooling water system capability following a design basis seismic event. There are no new failure modes or mechanisms created through use of the proposed evaluation methodology or pipe stress analysis with the proposed acceptance criteria. Use of these licensing basis changes with the cooling water system does not involve any modification in the operational limits of plant systems. There are no new accident precursors generated with use of these licensing basis changes.

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

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

Response: No.

This license amendment proposes to revise the plant licensing basis to include: (1) A hydraulic analysis methodology for demonstrating functionality of the CL system following a design basis seismic event; and (2) American Society of Mechanical Engineers, Section III, Subsection ND, ''Class 3 Components,'' 1986 Edition, Service Level D as the basis for acceptance criteria when performing stress analysis of the cooling water system non-Class I piping with design basis seismic loads.

The current plant licensing basis does not provide a hydraulic analysis methodology for demonstrating functionality of the cooling water system following a design basis seismic event and it does not provide acceptance criteria for piping stress analysis of the cooling water system non-Class I piping with design basis seismic loads. The proposed changes provide a conservative basis for evaluating cooling water system capability during and following a design basis seismic event. The proposed methodology for evaluating cooling water system capability is consistent with methods proposed by the NRC Staff and current plant methods for evaluating internal flooding. The intended use of the proposed acceptance criteria is consistent with the intended post-seismic use of the non-Class I portions of the cooling water system.

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

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

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

NRC Section Chief: L. Raghavan.

Omaha Public Power District, Docket No. 50–285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: May 14, 2004.

Description of amendment request: The proposed amendment revises the Fort Calhoun Station, Unit No. 1 (FCS) Technical Specifications to provide a risk-informed alternative to the existing restoration period for the high pressure safety injection (HPSI) system. The FCS application of the risk-informed change integrates the Westinghouse Owners Group recommendations identified in WCAP–15773, ''Joint Application Report for the Implementation of a Risk Management Technical Specification for the High Pressure Safety Injection (HPSI) System.''

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

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

The proposed change does not require any physical change to any plant systems, structures, or components nor does it require any change in systems or plant operations; thus the probability of an accident previously evaluated occurring remains unchanged. The proposed change does not require any change in safety analysis methods or results. A single HPSI subsystem inoperability is considered in existing plant analyses and regulatory criteria with respect to single failure criteria and the risk of extended HPSI subsystem outages are assessed in accordance with the Maintenance Rule [10 CFR 50.65]. Because risk is appropriately managed and compensatory measures established where necessary, the consequences of an accident previously analyzed are not significantly increased. The change to establish the extended HPSI CT [completion time] limits is justified because operation within the requirements of the Maintenance Rule continues to be governed by the current regulation and plant programs.

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.

HPSI System inoperabilities are assumed in existing analyses with respect to single failure criteria and are limited by existing regulation. Extending the time in which a HPSI component may remain inoperable does not constitute a change that could result in a new type of accident initiator than that previously identified. In addition, overall plant risk will be managed in accordance with the Maintenance Rule to help ensure continued safe plant operation.

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 require any change in accident analysis methods or results. Overall plant risks will continue to be appropriately managed and compensatory measures established when appropriate to reduce the overall risk during extended HPSI CT periods. In addition, an evaluation of common cause failure and a determination of the flow capacity of remaining ECCS [emergency core cooling system] components will continue to be performed in relation to HPSI System inoperabilities. Although components important to safety have an impact on overall plant risk and may impact the overall margin to safety, the adverse impacts that are realized due to single HPSI subsystem inoperabilities is largely offset by the avoidance of unnecessary shutdown transition risks and the establishment of compensatory measures and contingency actions where appropriate.

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: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: May 11, 2004.

Description of amendment request: The proposed amendment would revise the standby liquid control (SLC) pump discharge pressure surveillance requirement 3.1.7.7 acceptance criteria from 1224 psig to 1395 psig in the SSES 1 and 2 Technical Specification 3.1.7.

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

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

No. The proposed change establishes the operability requirements for the SLC subsystem based on its functional capability to operate during an ATWS [anticipated transients without scram] event. This proposed change to the surveillance for SLC pump discharge pressure does not affect the operation of any other SSES SSC's [structures, systems and components]. The SLC system is already being tested on a quarterly basis to the proposed new pump discharge pressure to demonstrate that the In Service Inspection Program requirements are met.

Consequently, the proposed change has no effect on the probability of any accident previously evaluated. Further, the consequences of any accident previously evaluated are not affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed change to the surveillance for SLC pump discharge pressure does not involve any physical alteration of the plant (no new or different type of equipment is installed) or changes in methods governing normal plant operation. Since this change does not introduce any new accident initiators, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The proposed change to the surveillance for SLC pump discharge pressure does not involve any physical alteration of the plant (no new or different type of equipment is installed) or changes in methods governing normal plant operation. The proposed change only affects determination of SLC system Technical Specification operability based on the functional capability of the SLC subsystems to inject boron during an ATWS event. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard J. Laufer.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: November 4, 2003.

Brief description of amendment: The amendment revises technical specification (TS) requirements for mode change limitations in Limiting Condition for Operation 3.0.4 and Surveillance Requirement 3.0.4 to adopt the provisions of Industry TS Task Force (TSTF) change TSTF–359, ''Increase Flexibility in Mode Restraints.''

Date of issuance: June 7, 2004.

Effective date: June 7, 2004, and shall be implemented within 60 days from the date of issuance.

Amendment No.: 187.

Facility Operating License No. NPF–21: The amendment revised the Technical Specifications.

*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 June 7, 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: February 3, 2004.

Brief description of amendment: The amendment revises Technical Specification 3.1.8, ''Scram Discharge Volume (SDV) Vent and Drain Valves,'' for the condition of having one or more SDV vent or drain lines with one valve inoperable.

Date of issuance: June 16, 2004.

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

Amendment No.: 140.

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

*Date of initial notice in **Federal Register**:* March 30, 2004 (69 FR 16619).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 16, 2004.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: October 21, 2003, as supplemented on March 1, 2004.

Brief description of amendment: The amendment revises Technical Specification 5.5.7.b.1 regarding the maximum time interval between steam generator (SG) inspections. The amendment permits, on a one-time basis, the extension of the SG inspection interval such that the next SG inspection, which would have been required to be performed no later than November 17, 2004, to be deferred until June 17, 2006. This effectively extends the current inspection interval from a maximum of 24 calendar months to 43 calendar months.

Date of issuance: June 23, 2004.

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

Amendment No.: 239.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* December 9, 2003 (68 FR 68663).

The supplement dated March 31, 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 June 23, 2004.

No significant hazards consideration comments received: No.

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

Date of amendment request: March 15, 2004.

Brief description of amendment: The amendment relocates the Waterford Steam Electric Station, Unit 3 Technical Specification 3.4.8.2, pressurizer heatup and cooldown limits, the associated action statements and surveillance requirements to the Technical Requirements Manual.

Date of issuance: June 16, 2004.

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

Amendment No.: 195.

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

*Date of initial notice in **Federal Register**:* April 13, 2004 (69 FR 19569).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 16, 2004.

No significant hazards consideration comments received: No.

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

Date of amendment request: October 22, 2003.

Brief description of amendment: The licensee is changing the existing pressure/temperature limits from 16 effective full power years (EFPY) to 32 EFPYs. In addition, the reactor coolant system maximum heatup and cooldown temperatures are changed to 60 °F and 100 °F/hour, respectively. For inservice hydrostatic pressure and leak testing, the maximum heatup and cooldown rates are now 60 °F and 100 °F respectively.

Date of issuance: June 16, 2004.

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

Amendment No.: 196.

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

*Date of initial notice in **Federal Register***: December 9, 2003 (68 FR 68667).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 16, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 18, 2004, and supplemented by letter dated June 8, 2004.

Brief description of amendment: The amendment deletes the requirements from the Technical Specifications to maintain hydrogen recombiners and hydrogen analyzers.

Date of issuance: June 16, 2004.

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

Amendment No: 166.

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

Date of initial notice in ***Federal Register***: March 16, 2004 (69 FR 12366).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 16, 2004.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket No. STN 50–454, Byron Station, Unit No. 1, Ogle County, Illinois

Date of application for amendment: December 5, 2003.

Brief description of amendment: The amendment permits a change in the fuel rod-average-burnup limit from 60,000 MWD/MTU to 65,000 MWD/MTU for four lead test assemblies during Byron Station, Unit 1, Cycle 13.

Date of issuance: June 16, 2004.

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

Amendment No.: 137.

Facility Operating License No. NPF–37: The amendment revised the License.

Date of initial notice in ***Federal Register***: January 20, 2004 (69 FR 2742).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 16, 2004.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Docket No. 50–277, Peach Bottom Atomic Power Station, Unit 2,York and Lancaster Counties, Pennsylvania

Date of application for amendments: February 12, 2004, as supplemented March 29, 2004.

Brief description of amendment: This amendment revised Technical Specification (TS) Table 3.3.6.1–1, ''Primary Containment Isolation Instrumentation,'' to increase the TS Allowable Value related to the setpoint for the Main Steam Tunnel Temperature—High system isolation function for those instruments located within the Reactor Building. A new Function, 1.f, has been added to represent the Reactor Building Main Steam Tunnel Temperature—High. Function 1.e has been renamed to clarify that it represents only the Turbine Building Main Steam Tunnel Temperature—High.

Date of issuance: June 16, 2004.

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

Amendment No.: 250.

Renewed Facility Operating License No. DPR–44: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: *March 2, 2004 (69 FR 9860).*

The March 29, 2004, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 16, 2004.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: January 30, 2004.

Brief description of amendment: The amendment deletes the Technical Specification requirements associated with the hydrogen and oxygen monitors.

Date of issuance: June 10, 2004.

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

Amendment No.: 254.

Facility Operating License No. DPR–49: The amendment revised the Technical Specifications.

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 10, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 8, 2003, as supplemented February 27 and May 3, 2004.

Brief description of amendment: The amendment revises the Technical Specifications with a one-time change to allow a 40-month inspection interval after the first (post-replacement) steam generator inservice inspection, rather than after two consecutive inspections resulting in a C–1 classification.

Date of issuance: June 18, 2004.

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

Amendment No.: 175.

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

Date of initial notice in ***Federal Register****:* January 20, 2004 (69 FR 2743).

The supplements dated February 27 and May 3, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 18, 2004.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan

Date of application for amendment: January 29, 2004, as supplemented on May 14, and June 2, 2004.

Brief description of amendment: The amendment grants approval to update the final safety analysis report (FSAR) to reflect the fuel pool building crane (L–3 crane) main hoist upgrade to the new rated capacity of 110 tons and reflect the new single-failure-proof design. Specifically, the amendment approves the use of the L–3 crane as a single-failure-proof crane for below-the-hook loads up to 110 tons.

Date of issuance: June 16, 2004.

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

Amendment No.: 215.

Facility Operating License No. DPR–20. Amendment updated the FSAR.

Date of initial notice in ***Federal Register****:* March 1, 2004 (69 FR 9649).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 16, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 15, 2003.

Brief description of amendments: The amendments revise Technical Specification (TS) 2.1.1.2 of TS Section 2.0, ''Safety Limits (SLs).'' The amendments replace the peak linear heat rate SL with a peak fuel centerline temperature SL so that the SL in TS 2.1.2.2 adequately conforms to 10 CFR 50.36(c)(1)(ii)(A) which requires that limiting safety system settings prevent a SL from being exceeded.

Date of issuance: June 10, 2004.

Effective date: June 10, 2004, to be implemented within 60 days of issuance.

Amendment Nos.: Unit 2–192 ; Unit 3–183.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register****:* October 14, 2003 (68 FR 59219).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 10, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 22, 2003 (TS 03–12).

Brief description of amendment: The amendments extend from 1 hour to 24 hours the completion time for Condition B of Technical Specification 3.5.1.1, which defines requirements for the restoration of an emergency core cooling system accumulator when it has been declared inoperable for a reason other than boron concentration.

Date of issuance: June 18, 2004.

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

Amendment Nos.: 291 and 281.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the technical specifications.

Date of initial notice in ***Federal Register***: January 6, 2004 (69 FR 699).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 6, 2003, as supplemented by the letter dated December 19, 2003.

Brief description of amendment: The amendment revises several surveillance requirements (SRs) in Technical Specifications (TSs) 3.8.1 and 3.8.4 on alternating current and direct current sources, respectively, for plant operation. The revised SRs have notes deleted or modified to allow the SRs to be performed, or partially performed, in reactor modes that previously were not allowed by the TSs. The licensee withdrew the changes to SRs 3.8.4.7 and 3.8.4.8 in its letter dated April 14, 2004.

Date of issuance: June 14, 2004.

Effective date: June 14, 2004, and shall be implemented within 60 days of the date of issuance including the incorporation of the changes to the Technical Specification Bases as described in the licensee's letters dated June 6 and December 19, 2003, and April 14, 2004.

Amendment No.: 162.

Facility Operating License No. NPF–30: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: July 22, 2003 (68 FR 43394).

The December 19, 2003, and April 14, 2004, supplemental letters provided additional clarifying information, did not expand the scope of the application as 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 June 14, 2004.

No significant hazards consideration comments received: No.

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

Date of amendment request: February 9, 2004.

Brief description of amendment: The amendment revises TS 5.5.7, ''Reactor Coolant Pump Flywheel Inspection Program,'' to increase the inspection interval from 10 years to 20 years.

Date of issuance: June 16, 2004.

Effective date: June 16, 2004, and shall be implemented within 90 days from the date of issuance.

Amendment No.: 153.

Facility Operating License No. NPF–42. The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: March 16, 2004 (69 FR 12373).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 16, 2004.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 28th day of June 2004.

For the Nuclear Regulatory Commission.

Edwin M. Hackett,

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

[FR Doc. 04–15061 Filed 7–2–04; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

Regulatory Guide; Issuance, Availability

The U.S. Nuclear Regulatory Commission (NRC) has issued a revision to a guide in its Regulatory Guide Series. This series has been developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques used by the staff in its review of applications for permits and licenses, and data needed by the NRC staff in its review of applications for permits and licenses.

The NRC has issued Revision 1 to Regulatory Guide 3.69, ''Topical Guidelines for the Licensing Support Network,'' which provides guidance acceptable to NRC Staff regarding the scope of documentary material that should be identified in or made available via the Licensing Support Network (LSN). The original version of this regulatory guide was published on September 19, 1996 (61 FR 49363). The LSN is an electronic information system that makes relevant documentary material available (via the Internet at *http://www.lsnnet.gov*) to parties, potential parties, and interested governmental participants in the adjudicatory proceeding on an application for a license to receive and possess high-level radioactive waste at a geologic repository operations area. The LSN facilitates document discovery similar to that available in NRC licensing proceedings. A proposed draft revision 1 of Regulatory Guide 3.69 (DG–3022) was made available for comment on July 2, 2002 (67 FR 44478). The proposed revision modified the topical guidelines to be consistent with the license application content specified in 10 CFR Part 63, ''Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada,'' (66 FR 55732, November 2, 2001), the structure of proposed Revision 2 of the ''Yucca Mountain Review Plan,'' NUREG–1804, published for comment on March 29, 2002 (67 FR 15257), the topics in the U.S. Department of Energy's ''Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada,'' dated February 2002, and the topics in Draft NUREG–1748, ''Environmental Review Guidance for Licensing Actions Associated with NMSS Programs,'' dated August 2001. The comment period for proposed Revision 1 of Regulatory Guide 3.69 (DG–3022) closed September 30, 2002.

This revision also reflects modifications made in response to comments and a recently issued change to 10 CFR 2.1005, which excludes ''Correspondence between a potential party, interested governmental participant, or party and the Congress of the United States'' from documentary material to be identified in or made available via the LSN. See ''Licensing Proceeding for a High-Level Radioactive Waste Geologic Repository; Licensing Support Network, Submissions to the