accommodate the scheduling priorities of the key participants.

R. Andrew Falcon,

Advisory Committee Management Officer, National Aeronautics and Space Administration.

[FR Doc. 04–10660 Filed 5–10–04; 8:45 am]

BILLING CODE 7510–01–P

NUCLEAR REGULATORY COMMISSION

[Docket Number 030–31768]

Notice of Availability of Environmental Assessment and Finding of No Significant Impact for License Amendment for Truman State University, Kirksville, MO

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability of Environmental Assessment and Finding of No Significant Impact.

FOR FURTHER INFORMATION CONTACT: Dr. Peter J. Lee, Division of Nuclear Materials Safety, U.S. Nuclear Regulatory Commission, Region III, 2443 Warrenville Road, Lisle, Illinois 60532–4352; telephone (630) 829–9870; or by email at *pjl2@nrc.gov.*

SUPPLEMENTARY INFORMATION:

I. Introduction

The U.S. Nuclear Regulatory Commission (NRC) is considering the issuance of a license amendment of Material License No. 24–17224–02 issued to Truman State University (the licensee), to terminate its license and authorize release of its Kirksville, Missouri facility for unrestricted use.

The NRC staff has prepared an Environmental Assessment (EA) in support of this licensing action in accordance with the requirements of 10 CFR Part 51. Based on the EA, the NRC has concluded that a Finding of No Significant Impact (FONSI) is appropriate. The amendment will be issued following the publication of this Notice.

II. EA Summary

The purpose of the proposed action is to terminate Truman State University's license and release its Kirksville, Missouri facility for unrestricted use. On July 25, 1990, the NRC authorized Truman State University to use labeled compounds of P–32, I–125, H–3, C–14, etc. for research and development. On December 18, 2003, Truman State University submitted a license amendment request to terminate its license and release its Kirksville facility for unrestricted use. Truman State University has conducted surveys of the facility and provided information to the NRC to demonstrate that the site meets the license termination criteria in Subpart E of 10 CFR Part 20 for unrestricted release. The staff has examined Truman State University's request and the information that the licensee has provided in support of its request, including the surveys performed by Truman State University to demonstrate compliance with 10 CFR 20.1402, '''Radiological Criteria for Unrestricted Use,''' to ensure that the NRC's decision is protective of the public health and safety and the environment.

III. Finding of No Significant Impact

The staff has prepared the EA (summarized above) in support of Truman State University's proposed license amendment to terminate its license and release the Kirksville facility for unrestricted use. Based on its review, the staff has determined that the affected environment and the environmental impacts associated with the decommissioning of Truman State University's facility are bounded by the impacts evaluated by the ''Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities'' (NUREG–1496). No outdoor areas were affected by the use of licensed materials. Additionally, no non-radiological impacts or other activities that could result in cumulative impacts were identified. The staff also finds that the proposed release for unrestricted use of the Truman State University's facility is in compliance with 10 CFR 20.1402. On the basis of the EA, the staff has concluded that the environmental impacts from the proposed action would not be significant. Accordingly, the staff has determined that a FONSI is appropriate, and has determined that the preparation of an environmental impact statement is not warranted.

IV. Further Information

In accordance with 10 CFR 2.390 of the NRC's ''Rules of Practice,'' Truman State University's request, the EA summarized above, and the documents related to this proposed action are available electronically for public inspection and copying from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html.* These documents include Truman State University's NRC Form 314 dated December 18, 2003, with enclosures (Accession No. ML041120082); and the EA summarized above (Accession No. ML041190131). These documents may also be viewed electronically on the public computers located at the NRC's Public Document Room (PDR), O 1 F21, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852. The PDR reproduction contractor will copy documents for a fee. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1–800–397–4209 or (301) 415–4737, or by e-mail to *pdr@nrc.gov.*

Dated at Lisle, Illinois, this 28th day of April, 2004.

William G. Snell,

Acting Chief, Decommissioning Branch, Division of Nuclear Materials Safety, RIII.

[FR Doc. 04–10614 Filed 5–10–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Biweekly Notice; Applications and Amendments to Facility Operating Licenses

Involving No Significant Hazards Considerations

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from, April 16 through April 29, 2004. The last biweekly notice was published on April 27, 2004 (69 FR 22877).

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

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

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

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

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed by: (1) First class mail addressed to the

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

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

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

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

Date of amendment request: March 23, 2004.

Description of amendment request: The licensee requested to revise the Technical Specifications (TSs), deleting the requirements for the Independent Onsite Safety Review Group (IOSRG) and locating them intact to a licensee-controlled document, the company-wide Quality Assurance Topical Report (QATR). The requirements are in the administrative section of the TSs and include IOSRG organization, function description, member qualifications, and recordkeeping. The relocation is proposed per the guidance of Nuclear Regulatory Commission (NRC) Administrative Letter 95–06. In addition, the licensee proposed to correct the reference for facility activities audits from a site-specific document to the company-wide QATR.

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 reviewed the licensee's analysis and has performed its own analysis as follows:

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

No. The proposed amendment does not affect assumptions contained in the current licensing basis plant safety analyses, will not lead to physical changes of a plant structure, system, or component (SSC), and will not alter the method of operation of any SSC. The IOSRG requirements and conduct of IOSRG activities were not factors in any previously analyzed accident or transient scenarios, and thus, the elimination of IOSRG requirements from the TSs will have no effect on the probability of occurrence and consequences of any previously analyzed accident or transient.

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

No. The proposed amendment is not the result of a design change or method of operation change, and will not lead to such changes. Hence no, new or different kind of accident can be created from any accident previously evaluated.

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

No. The proposed amendment does not involve any change to current analysis models, assumptions, limiting conditions for operation, operational parameters, action statements, and surveillance requirements. Hence, there is no reduction in a margin of safety.

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

Attorney for licensee: Edward J. Cullen, Jr., Esquire, Vice President, General Counsel and Secretary, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: March 23, 2004.

Description of amendment request: The licensee requested to revise the Technical Specifications (TSs), deleting the requirements for the Independent Onsite Safety Review Group (IOSRG) and locating them intact to a licensee-controlled document, the company-wide Quality Assurance Topical Report (QATR). The requirements are in the administrative section of the TSs and include IOSRG organization, function description, member qualifications, and recordkeeping. The relocation is proposed per the guidance of Nuclear Regulatory Commission (NRC) Administrative Letter 95–06. In addition, the licensee proposed to correct the reference for facility activities audits from a site-specific document to the company-wide QATR.

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 reviewed the licensee's analysis and has performed its own analysis as follows:

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

No. The proposed amendment does not affect assumptions contained in the current licensing basis plant safety analyses, will not lead to physical changes of a plant structure, system, or component (SSC), and will not alter the method of operation of any SSC. The IOSRG requirements and conduct of IOSRG activities were not factors in any previously analyzed accident or transient scenarios, and thus, the elimination of IOSRG requirements from the TSs will have no effect on the probability of occurrence and consequences of any previously analyzed accident or transient.

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

No. The proposed amendment is not the result of a design change or method of operation change, and will not lead to such changes. Hence, no new or different kind of accident can be created from any accident previously evaluated.

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

No. The proposed amendment does not involve any change to current analysis models, assumptions, limiting conditions for operation, operational parameters, action statements, and surveillance requirements. Hence, there is no reduction in a margin of safety.

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

Attorney for licensee: Edward J. Cullen, Jr., Esquire, Vice President, General Counsel and Secretary, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348

NRC Section Chief: Richard J. Laufer.

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: March 4, 2004.

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

The NRC staff issued a notice of availability of a model no significant hazards consideration determination for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated March 4, 2004.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: April 15, 2004.

Description of amendment request: The proposed amendment would change the reactor coolant system (RCS) pressure/temperature (P/T) limits in the technical specifications (TSs) by providing a single maximum cooldown rate instead of a variable cooldown rate and by revising the cooldown curve with one that is slightly more restrictive.

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 probability of occurrence of an accident previously evaluated for ANO–2 [Arkansas Nuclear One, Unit 2] is not altered by the proposed amendment to the TSs. The accidents remain the same as currently analyzed in the ANO–2 Safety Analysis Report (SAR) as a result of the change to the cooldown P/T limits. The new P/T cooldown limits were based on NRC [Nuclear Regulatory Commission] accepted methodologies along with ASME [American Society of Mechanical Engineers] Code [Boiler and Pressure Vessel Code] alternatives. The proposed change does not impact the integrity of the reactor coolant pressure boundary (RCPB) (i.e., there is no change to the operating pressure, materials, loadings, etc.) as a result of this change. In addition, there is no increase in the potential for the occurrence of a loss of coolant accident. The proposed P/T cooldown limit curve is not considered to be an initiator or contributor to any accident currently evaluated in the ANO–2 SAR. The revised P/T cooldown limits ensure the long term integrity of the RCPB. For each analyzed transient and steady state condition, the allowable pressure was determined as a function of reactor coolant temperature considering postulated flaws in the reactor vessel beltline, inlet nozzle, outlet nozzle, and closure head flange.

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 to the P/T limits will not create a new accident scenario. The requirements to have P/T protection are part of the ANO–2 licensing basis. The proposed change in the P/T cooldown limits is based on NRC approved methodologies performed by Framatome ANP. This methodology complies with NRC and ASME requirements for protecting the RCS. Therefore, the revised P/T cooldown limits provide protection of the RCS from limiting transients during normal cooldown.

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 revision of the P/T limits and curves will ensure that ANO–2 continues to operate within the operating margins of the ASME Code. The application of ASME Code Cases N–640 and N–588 presents alternative procedures for calculating P/T temperatures and pressures. These Code Cases allow certain assumptions to be conservatively reduced. However, the procedures allowed by these Code Cases still provide sufficient conservatism and ensure an adequate margin of safety in the development of P/T operating and pressure test limits to prevent non-ductile fractures.

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

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

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

NRC Section Chief: Robert A. Gramm.

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: This license amendment request (LAR) proposes to eliminate the Technical Specification Surveillance Requirements (SRs) that require each Main Steam Safety/Relief Valve (S/RV) to open during the manual actuation portion of testing the valves. In accordance with 10 CFR 50.55a, ''Codes and Standards,'' paragraph (a)(3), this request also includes Relief Request VR–13. VR–13 is a request for relief from the requirements of ASME/American National Standards Institute (ANSI) Operation and Maintenance (OM) of Nuclear Power Plants, OM–1995, Appendix I, Section 3.4.1(d) that after isolation, the S/RVs are manually opened and closed.

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 [License Amendment Request] LAR modifies TS 3.4.4.3, SR 3.5.1.7, and SR 3.6.1.6.1 to allow the uncoupling of the S/RV stem from the S/RV actuator during manual actuation. The proposed LAR does not change the manner in which the S/RVs are intended to operate.

The performance of S/RV testing provides assurance that the S/RVs are capable of depressurizing the Reactor Pressure Vessel (RPV). This will protect the RPV from over pressurization and allows the combination of the Low Pressure Coolant Injection (LPCI) system and the Low Pressure Core Spray (LPCS) system to inject into the RPV as designed. The proposed testing requirements are sufficient to provide confidence that the S/RVs, [Automatic Depressurization System] ADS valves, and the [Low-Low Set] LLS valves will perform their intended design safety functions.

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

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

The proposed LAR changes TS [Surveillance Requirements] SR 3.4.4.3, SR 3.5.1.7, and SR 3.6.1.6.1. The changes to these SRs do not effect the assumed accident performance of the S/RVs, nor any plant structure, system or component previously evaluated. The LAR does not install any new equipment, nor does it cause existing equipment to be operated in a new or different manner. The S/RVs continue to be bench-tested to verify the safety and relief modes of valve operation. The changes will allow the testing of the manual actuation electrical circuitry, solenoid and air control valve, and the actuator without causing the S/RV to open. No setpoints are being changed which would alter the dynamic response of plant equipment.

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

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

The proposed LAR will allow the uncoupling of the S/RV stem from the other components associated with the manual actuation testing of the S/RVs. The proposed changes will allow the testing of the manual actuation electrical circuitry, solenoid and air control valve, and the actuator without causing the S/RV to open. The S/RVs will continue to be manually actuated by the bench-test of the valve control system and setpoint testing program prior to installation in the plant. The changes do not effect the valve setpoint or operational criteria that directs the S/RVs to be manually opened during plant transients. There are no changes which alter the setpoints at which protective actions are initiated.

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

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: April 5, 2004.

Description of amendment request: This license amendment request (LAR) proposes to modify the existing Minimum Critical Power Ratio (MCPR) Safety Limit contained in Technical Specification 2.1.1.2. Specifically, the change modifies the MCPR Safety Limit values, as calculated by Global Nuclear Fuel (GNF), by decreasing the limit for two recirculating loop operation from 1.10 to 1.08, and decreasing the limit for single recirculation loop operation from 1.11 to 1.10. The change resulted from a core reload analysis performed during the PNPP Fuel Cycle 10.

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.

Perry Nuclear Power Plant (PNPP) Updated Safety Analysis Report (USAR) Section 4.2, ''Fuel System Design,'' states the PNPP fuel system design bases are provided in the General Electric Topical Report, NEDE–24011–P–A, ''General Electric Standard Application for Reactor Fuel (GESTAR II).'' The Minimum Critical Power Ratio (MCPR) Safety Limit is one of the limits used to protect the fuel in accordance with the design basis. The MCPR Safety Limit establishes a margin to the onset of transition boiling. The basis of the MCPR Safety Limit remains the same, ensuring that greater than 99.9 % of all fuel rods in the core avoid transition boiling. The methodology used to determine the MCPR Safety Limit values is contained within GESTAR II and is NRC approved. The change does not result in any physical plant modifications or physically affect any plant components. As a result, there is no increase in the probability of occurrence of a previously analyzed accident.

The fundamental sequences of accidents and transients have not been altered. The Safety Limit MCPR is established to avoid fuel damage in response to anticipated operational occurrences. Compliance with a MCPR Safety Limit greater than or equal to the calculated value will ensure that less than 0.1% of the fuel rods will experience boiling transition. This in turn ensures fuel damage does not occur following transients due to excessive thermal stresses on the fuel cladding. The MCPR Operating Limits are set higher (i.e., more conservative) than the Safety Limit such that potentially limiting plant transients prevent the MCPR from decreasing below the MCPR Safety Limit during the transient. Therefore, there is no impact on any of the limiting USAR Appendix 15B transients. The radiological consequences remain the same as previously stated in the USAR. Therefore, the consequences of an accident do not increase over previous evaluations in the USAR.

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

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

The MCPR Safety Limit basis is preserved, which is to ensure that transition boiling does not occur in at least 99% of the fuel rods in the core as a result of the postulated limiting transient. The values are calculated in accordance with GESTAR II. The GESTAR II analyses have been accepted by the NRC. The MCPR Safety Limit is one of the limits established to ensure the fuel is protected in accordance with the design basis. The function, location, operation, and handling of the fuel remain unchanged. No changes in the design of the plant or the method of operating the plant are associated with these revised safety limit valves. Therefore, no new or different kind of accident from any previously evaluated is created.

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

This change revises the PNPP MCPR Safety Limit values. The new MCPR Safety Limit values reflect changes due to Cycle 10 core design, but do not alter the design or function of any plant system, including the fuel. The new MCPR Safety Limit values were calculated using NRC-approved methods described in GESTAR II. The proposed MCPR Safety Limit values continue to satisfy the fuel design safety criteria which ensures that transition boiling does not occur in at least 99.9% of the fuel rods in the core as a result of the postulated limiting transient. Therefore, the proposed values for the MCPR Safety Limit do not involve a significant reduction in a safety margin.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: April 5, 2004.

Description of amendment request: This license amendment request (LAR) proposes to modify the existing Minimum Critical Power Ratio (MCPR) Safety Limit contained in Technical Specification 2.1.1.2. Specifically, the change modifies the MCPR Safety Limit values, as calculated by Global Nuclear Fuel (GNF), by decreasing the limit for two recirculating loop operation from 1.10 to 1.08, and decreasing the limit for single recirculation loop operation from 1.11 to 1.10. The change resulted from a core reload analysis performed during the PNPP Fuel Cycle 10.

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.

Perry Nuclear Power Plant (PNPP) Updated Safety Analysis Report (USAR) Section 4.2, ''Fuel System Design,'' states the PNPP fuel system design bases are provided in the General Electric Topical Report, NEDE–

24011–P–A, ''General Electric Standard Application for Reactor Fuel (GESTAR II).'' The Minimum Critical Power Ratio (MCPR) Safety Limit is one of the limits used to protect the fuel in accordance with the design basis. The MCPR Safety Limit establishes a margin to the onset of transition boiling. The basis of the MCPR Safety Limit remains the same, ensuring that greater than 99.9 % of all fuel rods in the core avoid transition boiling. The methodology used to determine the MCPR Safety Limit values is contained within GESTAR II and is NRC approved. The change does not result in any physical plant modifications or physically affect any plant components. As a result, there is no increase in the probability of occurrence of a previously analyzed accident.

The fundamental sequences of accidents and transients have not been altered. The Safety Limit MCPR is established to avoid fuel damage in response to anticipated operational occurrences. Compliance with a MCPR Safety Limit greater than or equal to the calculated value will ensure that less than 0.1% of the fuel rods will experience boiling transition. This in turn ensures fuel damage does not occur following transients due to excessive thermal stresses on the fuel cladding. The MCPR Operating Limits are set higher (i.e., more conservative) than the Safety Limit such that potentially limiting plant transients prevent the MCPR from decreasing below the MCPR Safety Limit during the transient. Therefore, there is no impact on any of the limiting USAR Appendix 15B transients. The radiological consequences remain the same as previously stated in the USAR. Therefore, the consequences of an accident do not increase over previous evaluations in the USAR.

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

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

The MCPR Safety Limit basis is preserved, which is to ensure that transition boiling does not occur in at least 99% of the fuel rods in the core as a result of the postulated limiting transient. The values are calculated in accordance with GESTAR II. The GESTAR II analyses have been accepted by the NRC. The MCPR Safety Limit is one of the limits established to ensure the fuel is protected in accordance with the design basis. The function, location, operation, and handling of the fuel remain unchanged. No changes in the design of the plant or the method of operating the plant are associated with these revised safety limit valves. Therefore, no new or different kind of accident from any previously evaluated is created.

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

This change revises the PNPP MCPR Safety Limit values. The new MCPR Safety Limit values reflect changes due to Cycle 10 core design, but do not alter the design or function of any plant system, including the fuel. The new MCPR Safety Limit values were calculated using NRC-approved methods described in GESTAR II. The proposed MCPR Safety Limit values continue to satisfy the fuel design safety criteria which ensures that transition boiling does not occur in at least 99.9% of the fuel rods in the core as a result of the postulated limiting transient. Therefore, the proposed values for the MCPR Safety Limit do not involve a significant reduction in a safety margin.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment requests: August 27, 2003.

Description of amendment requests: The proposed amendments would amend Unit 1 and Unit 2 Technical Specifications (TS) 4.0.3. TS 4.0.3 describes the relationship between meeting the surveillance requirement and operability. The proposed change will modify TS 4.0.3 to allow a missed surveillance to be completed within 24 hours or up to the limit of the specified interval, whichever is greater. Additionally, a statement that a risk evaluation shall be performed for any surveillance delayed greater than 24 hours and that the risk impact shall be managed is being added to the TS.

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 relaxes the time allowed to perform a missed surveillance. The time between surveillances is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Any reduction in confidence that a standby system might fail to perform its safety function due to a missed surveillance is small and would not, in the absence of other unrelated failures, lead to an increase in consequences beyond those estimated by existing analyses. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. A missed surveillance will not, in and of itself, introduce new failure modes or effects and any increased chance that a standby system might fail to perform its safety function due to a missed surveillance would not, in the absence of other unrelated failures, lead to an accident beyond those previously evaluated. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. The format changes are intended to improve readability and appearance and do not alter any requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No.

The extended time allowed to perform a missed surveillance does not result in a significant reduction in the margin of safety. As supported by the historical data, the likely outcome of any surveillance is verification that the limiting condition for operation is met. Failure to perform a surveillance within the prescribed frequency does not cause equipment to become inoperable. The only effect of the additional time allowed to perform a missed surveillance on the margin of safety is the extension of the time until inoperable equipment is discovered to be inoperable by the missed surveillance. However, given the rare occurrence of inoperable equipment, and the rare occurrence of a missed surveillance, a missed surveillance on inoperable equipment would be very unlikely. This must be balanced against the real risk of manipulating the plant equipment or condition to perform the missed surveillance. In addition, parallel trains and alternate equipment are typically available to perform the safety function of the equipment not tested. Thus, there is confidence that the equipment can perform its assumed safety function. The format changes are intended to improve readability and appearance and do not alter any requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: L. Raghavan.

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

Date of amendment requests: February 14, 2004.

Description of amendment requests: The proposed amendments would revise the Technical Specifications (TS) governing containment penetrations and the Containment Purge and Exhaust Isolation System, which are applicable during CORE ALTERATIONS and movement of irradiated fuel, such that those TSs are only applicable during the movement of recently irradiated fuel.

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

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

Response: No.

The proposed changes incorporate line item improvements that are based on assumptions in the postulated fuel handling accident (FHA) analysis. These proposed changes remove the applicability of the Technical Specifications (TS) governing containment penetrations and the Containment Purge and Exhaust Isolation System when handling fuel assemblies that have decayed for a sufficient period of time. The containment penetration and Containment Purge and Exhaust Isolation System are not initiators to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The only previously analyzed accident affected by the proposed change is an FHA. The current, Nuclear Regulatory Commission (NRC)-approved analysis of an FHA does not assume any holdup of the postulated radioactivity release by the containment building nor does it assume the operation of the Containment Purge and Exhaust Isolation System. As a result, the proposed change does not affect the assumed mitigation or consequences of that event.

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 incorporate line item improvements that are based on assumptions in the postulated FHA analysis. These proposed changes remove the applicability of the TS governing containment penetrations and the Containment Purge and Exhaust Isolation System when handling fuel assemblies that have decayed for a sufficient period of time. The proposed changes do not involve the addition or modification of equipment nor do they alter the design of the plant. The revised operations are consistent with the FHA analysis and do not require any new or different ways of operating the plant equipment.

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

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

Response: No.

The proposed changes incorporate line item improvements that are based on assumptions in the postulated FHA analysis. These proposed changes remove the applicability of the TS governing containment penetrations and the Containment Purge and Exhaust Isolation System when handling fuel assemblies that have decayed for a sufficient period of time. The calculated offsite and Control Room doses resulting from an FHA are not affected by this change as the proposed TS changes are revised to be consistent with the assumptions used in these analyses. As a further measure, [Indiana Michigan Power Company] I&M has committed to maintaining a single normal or contingency method to promptly close containment penetrations following an FHA. These prompt methods will enable the ventilation systems to draw the release from a postulated FHA such that it can be treated and monitored. This will provide a further margin of safety beyond that assumed in the accident analysis.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107

NRC Section Chief: L. Raghavan.

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

Date of amendment requests: February 14, 2004.

Description of amendment requests: The proposed amendments would modify the Technical Specification (TS) 3.9.2 limiting condition for operation, to delete TS Surveillance Requirements (SRs) 4.9.2.a and b for the Source Range Neutron Flux Monitor channel functional test, to revise SR 4.9.2.c for the channel check test, and to add a requirement to perform a channel calibration every 18 months as well as revise TS 4.10.4.2 and 4.10.3.2 (Units 1 and 2 respectively) for Intermediate and Power Range channel functional test.

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

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

Response: No.

The proposed amendment replaces the Technical Specification (TS) 3.9.2 limiting condition for operation (LCO) requirement for an audible indication in the containment (both units) and control room (Unit 2) with a requirement that a source range audible count rate circuit be operable. This involves no physical changes to the plant, and maintains the capability to alert the operators to changes in core reactivity. Thus, neither the probability of an accident nor the consequences are significantly increased.

The proposed amendment revises the TS SR for the Power Range, Intermediate Range, and the Source Range Neutron Flux Monitors to reduce redundant testing. Surveillance testing is not an initiator to any accident previously evaluated. As a result, the proposed changes will not result in a significant increase in the probability of any accident previously evaluated.

The Power Range, Intermediate Range, and the Source Range Neutron Flux Monitors are used to detect and mitigate accidents previously evaluated. However, the LCOs continue to require the subject flux monitors to be operable and the remaining testing is sufficient to ensure the flux monitors are capable of performing their detection and mitigation functions. Thus, the consequences of an accident are not significantly changed.

Based on the above, [Indiana Michigan Power Company] I&M concludes that 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 and accident previously evaluated?

Response: No.

The proposed amendment replaces the TS 3.9.2 LCO requirement for an audible indication in the containment (both units) and control room (Unit 2) with a requirement that a source range audible count rate circuit be operable.

The change does not make any physical changes to the plant. Thus, the change does not create the possibility of a new or different kind of accident.

The proposed amendment revises the TS SR for the Power Range, Intermediate Range, and the Source Range Neutron Flux Monitors to reduce redundant testing. The proposed changes do not change the design function or operation of any plant equipment. No new failure mechanisms, malfunctions, or accident initiators are being introduced by the proposed changes. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed amendment replaces the TS 3.9.2 LCO requirement for an audible indication in the containment (both units) and control room (Unit 2) with a requirement that a source range audible count rate circuit be operable. The source range audible count rate circuit will continue to perform its function of alerting the operators to changes in core reactivity.

The proposed amendment revises the TS Surveillance Requirement (SR) for the Power Range, Intermediate Range, and the Source Range Neutron Flux Monitors to reduce redundant testing. The elimination of redundant testing does not reduce the reliability of the tested flux monitors. The flux monitors continue to be tested in a manner and at a frequency necessary to provide confidence that the equipment can perform its assumed safety function.

Therefore, there is no 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: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107

NRC Section Chief: L. Raghavan.

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

Date of amendment requests: April 6, 2004.

Description of amendment requests: The proposed amendments would revise Technical Specification (TS) design features for fuel assemblies and new fuel storage criticality limitations. In addition, the licensee requests approval of the criticality analysis methodology supporting the spent fuel storage rack and new fuel storage rack in accordance with 10 CFR 50.59(c)(2)(viii).

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

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

Response: No.

The proposed Technical Specification (TS) changes allow the zirconium-based alloy, M5, to be used in addition to Zircaloy-4 and ZIRLO in Donald C. Cook Nuclear Plant fuel assemblies. TS changes are also proposed to allow Gadolinia to be used in fuel assemblies in the new fuel storage racks to ensure adequate reactivity margin. In addition, methodology changes were proposed for a criticality analysis supporting new and spent fuel rack design criteria. M5 is a Nuclear Regulatory Commission (NRC)-approved alloy for fuel cladding and Gadolinia is an NRC-approved fuel burnable absorber used in the maintenance of reactivity margin in the new fuel storage rack. The use of NRC-approved cladding and fuel absorbers and methodology changes to criticality analyses to support TS design criteria for the spent and new fuel storage racks are not initiators of any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. M5 cladding has been shown to meet all 10 CFR 50.46 acceptance criteria. Analysis has shown that the use of Gadolinia assures sufficient reactivity margin to prevent a criticality accident in the new fuel storage rack. Changes in methodology for criticality analyses were performed to demonstrate TS requirements are met or to support proposed TS changes and do not affect plant equipment. Therefore, the consequences of an accident are not significantly increased.

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

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

Response: No.

The proposed change to use the M5 alloy is based on an NRC-approved topical report which demonstrates that the material properties of the M5 alloy are not significantly different from those of Zircaloy-4. The design and performance criteria continue to be met and no new failure mechanisms have been identified. Therefore, M5 fuel rod cladding and fuel assembly structural components will perform similarly to those fabricated from Zircaloy-4, thus precluding the possibility of the fuel becoming an accident initiator and causing a new or different type of accident.

The proposed TS change to use Gadolinia to ensure adequate reactivity margin for higher enrichment fuel assemblies prevents reactivity limits from being exceeded. An NRC-approved topical report demonstrates that Gadolinia is acceptable for use in fuel assemblies. The proposed change only modifies the type of fuel burnable absorber and does not affect any permanent plant equipment or plant operating procedures, and can not be an initiator of an accident.

The proposed criticality analysis supports TS design criteria for spent and new fuel racks. The analysis evaluates reactivity margin based on conservative assumptions on fuel assembly design and burnup and does not affect any plant equipment. The criticality analysis can not be an initiator of an accident.

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 TS change to allow the use of fuel rods clad with the M5 alloy does not change the reactor fuel reload design and safety limits. For each cycle reload core, the fuel assembly design and core configuration are evaluated using NRC-approved reload design methods, including consideration of the core physics analysis peaking factors and core average linear heat rate effects. The design basis and modeling techniques for fuel assemblies with Zircaloy-4 and ZIRLO clad fuel rods remain valid for fuel assemblies with M5 clad fuel rods. Use of the M5 alloy as cladding material has no effect on the criticality analysis for the spent fuel storage racks and the new fuel storage racks. Furthermore, it has no effect on the thermal-hydraulic and structural analysis for the spent fuel pool. Therefore, the design and safety analysis limits specified in the TS are maintained with this proposed change.

The proposed TS change to use Gadolinia as a fuel burnable absorber for fuel assemblies with higher enrichments of Uranium-235 to ensure proper reactivity control in the spent fuel storage rack is consistent with the current method of reducing reactivity of high enrichment fuel assemblies. Each method reduces the equivalent uranium enrichment to below that found acceptable by the NRC for safe storage of new fuel.

The proposed criticality analyses use NRC-approved codes with a methodology different than previously approved by the NRC. The criticality analysis results for the spent fuel storage rack flooded with unborated water condition and for the new fuel storage rack moderated by aqueous foam condition remain less than the limiting TS values. Analysis results for the new fuel storage rack flooded with unborated water condition are consistent with previous analysis results.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: April 13, 2004.

Brief description of amendments: The requested amendments will revise the Technical Specification 3.3.2, ''Engineered Safety Features Actuation System (ESFAS) Instrumentation,'' to revise the trip setpoint allowable value for Refueling Water Storage Tank (RWST) Low-Low Level (ESFAS function 7.b) for Unit 2 to be the same as it is for Unit 1. Also, the frequency of calibration of the RWST water level transmitters will be revised from once in 9 months to once in 18 months.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration by focusing on the three standards set forth in 10 CFR 50.92. The licensee's analysis of three standards is presented below:

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

Response: No.

The proposed change in the trip setpoint allowable value for Unit 2 Refueling Water Storage Tank (RWST) Low-Low Level has no impact on the probability of any accident previously evaluated. Since none of the accident analyses are affected by this change, the consequences of all previously evaluated accidents remain unchanged.

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.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of these changes. There are no changes in the method by which any safety-related plant system performs its safety function. Overall protection system performance will remain within the bounds of the previously performed accident analyses and the protection systems will continue to function in a manner consistent with the plant design basis. The proposed changes do not affect the probability of any event initiators. The proposed changes do not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the Final Safety Analysis Report (FSAR).

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

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

Response: No.

The proposed changes do not affect the acceptance criteria for any analyzed event nor is there a change to any Safety Analysis Limit (SAL). There will be no effect on the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, the Departure from Nucleate Boiling Ratio (DNBR) limits, the Heat Flux Hot Channel Factor (F_Q), the Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$), the Loss of Coolant Accident Peak Centerline Temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: April 8, 2004.

Description of amendment request: The proposed amendment revises TS 5.5.7, ''Reactor Coolant Pump Flywheel Inspection Program,'' to extend the allowable inspection interval to 20 years.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on June 24, 2003 (68 FR 37590), on possible amendments to extend the inspection interval for reactor coolant pump (RCP) flywheels, 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 October 22, 2003, (68 FR 60422). The licensee affirmed the applicability of the model NSHC determination in its application dated April 8, 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 to the RCP flywheel examination frequency does not change the response of the plant to any accidents. The RCP will remain highly reliable and the proposed change will not result in a significant increase in the risk of plant operation. Given the extremely low failure probabilities for the RCP motor flywheel during normal and accident conditions, the extremely low probability of a loss-of-coolant accident (LOCA) with loss of offsite power (LOOP), and assuming a conditional core damage probability (CCDP) of 1.0 (complete failure of safety systems), the core damage frequency (CDF) and change in risk would still not exceed the NRC's acceptance guidelines continued in Regulatory Guide (RG) 1.174 (<1.0E–6 per year). Moreover, considering the uncertainties involved in this evaluation, the risk associated with the postulated failure of an RCP motor flywheel is significantly low. Even if all four RCP motor flywheels are considered in the bounding plant configuration case, the risk is still acceptably low.

The proposed change does not adversely affect accident initiators or precursors, nor alter the design assumptions, conditions, or configuration of the facility, or the manner in which the plant is operated and maintained; alter or prevent the ability of structures, systems, components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits; or affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed change does not increase the type or amount of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposure. The proposed change is consistent with the safety analysis assumptions and resultant consequences. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change in flywheel inspection frequency does not involve any change in the design or operation of the RCP. Nor does the change to examination frequency affect any existing accident scenarios, or create any new or different accident scenarios. Further, the change does not involve a physical alteration of the plant (*i.e.*, no new or different type of equipment will be installed) or alter the methods governing normal plant operation. In addition, the change does not impose any new or different requirements or eliminate any existing requirements, and does not alter any assumptions made in the safety analysis. The proposed change is consistent with the safety analysis assumptions and current plant operating practice. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The safety analysis acceptance criteria are not impacted by this change. The proposed change will not result in plant operation in a configuration outside of the design basis. The calculated impact on risk is insignificant and meets the acceptance criteria contained in RG 1.174. There are no significant mechanisms for inservice degradation of the RCP flywheel. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

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

Date of application request: April 8, 2004.

Description of amendment request: The proposed change allows entry into a mode or other specified condition in the applicability of a technical specification (TS), while in a condition statement and the associated required actions of the TS, provided the licensee performs a risk assessment and manages risk consistent with the program in place for complying with the requirements of Title 10 of the Code of Federal Regulations (10 CFR), Part 50, Section 50.65(a)(4). Limiting Condition for Operation (LCO) 3.0.4 exceptions in individual TSs would be eliminated, several notes or specific exceptions are revised or deleted to reflect the related changes to LCO 3.0.4, and Surveillance Requirement (SR) 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 April 8, 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: John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Stephen Dembek.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: December 20, 2002, as supplemented on May 30, September 10, and November 3, 2003.

Brief description of amendment: The amendment authorized the revision of the OCNGS Updated Final Safety Analysis Report (UFSAR) to reflect implementation of the Boiling Water Reactor Vessel and Internals Project reactor pressure vessel Integrated Surveillance Program (ISP) as the basis for demonstrating compliance with the requirements of Appendix H, ''Reactor Vessel Material Surveillance Program Requirements,'' to Title 10 of the *Code of Federal Regulations,* Part 50.

Date of Issuance: April 27, 2004.

Effective date: The amendment is effective immediately. The ISP shall be implemented prior to the next scheduled reactor vessel surveillance capsule removal. The UFSAR is to be revised to reflect use of the ISP in accordance with the schedule of 10 CFR 50.71(e).

Amendment No.: 242.

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

Date of initial notice in ***Federal Register:*** *February 4, 2003 (68 FR 5669).*

The May 30, September 10, and November 3, 2003, letters provided clarifying information within the scope of the original application, and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated April 27, 2004. No significant hazards consideration comments received: No.

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

Date of application for amendments: May 12, 2003, as supplemented December 5, 2003, February 23, 2004, March 26, 2004 and April 6, 2004.

Brief description of amendments: These amendments extend several Required Action completion times for inoperable diesel generators identified in Technical Specification 3.8.1, ''AC Sources Operating.''

Date of issuance: April 13, 2004.

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

Amendment Nos.: 265 and 242.

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

Date of initial notice in ***Federal Register:*** June 24, 2003 (68 FR 37576). The licensee's December 5, 2003, February 23, 2004, March 26, 2004, and April 6, 2004, letters provided additional information that clarified the application, did not change 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 these amendments is contained in a Safety Evaluation dated April 13, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 27, 2003.

Brief description of amendments: The amendments modified Technical Specification 4.0.5.f and associated Bases, and Bases Section 3/4.4.8, with regard to the commitment to perform piping inspections in accordance with Generic Letter 88–01, by adding the words ''or in accordance with alternate measures approved by the NRC staff.''

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

Effective date: April 20, 2004.

Amendment Nos.: 171 and 133.

Facility Operating License Nos. NPF–39 and NPF–85. The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 19, 2003 (68 FR 49817).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 20, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 30, 2003.

Brief description of amendment: By letter dated January 30, 2003, FirstEnergy Nuclear Operating Company, (FENOC), the licensee for Perry Nuclear Power Plant (PNPP), Unit 1, submitted a request for Nuclear Regulatory Commission review and approval of a license amendment to modify the basis for their compliance with the requirements of Appendix H to Title 10 Part 50 of the Code of Federal Regulations (Appendix H to 10 CFR Part 50), ''Reactor Vessel Material Surveillance Program Requirements.'' In the license amendment submittal, FENOC requested that they be approved to implement the Boiling Water Reactor Vessel and Internals Project reactor pressure vessel integrated surveillance program as the basis for demonstrating the compliance of PNPP, Unit 1, with the requirements of Appendix H to 10 CFR Part 50.

Date of issuance: April 15, 2004.

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

Amendment No.: 128.

Facility Operating License No. NPF–58: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 6, 2004 (69 FR 696).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 14, 2003.

Brief description of amendment: By letter dated January 14, 2003, FirstEnergy Nuclear Operating Company, the licensee for Perry Nuclear Power Plant, Unit 1, submitted a request for Nuclear Regulatory Commission review and approval of a license amendment to modify the Technical Specifications (TS) 5.1.1, 5.4.1, and 5.5.1 to replace the requirement for the plant manager to approve administrative procedures and the Offsite Dose Calculation Manual. The plant manager approval signature will be replaced with the signature of a procedurally authorized individual who would be the more appropriate authority for approval of the activity. Additionally, a change is proposed to revise License Condition 2.F, to replace the 30-day reporting period with a direct reference to the 10 CFR 50.73 subsection that contains the reporting period. The License Condition already references 10 CFR 50.73 for use in reporting plant issues.

Date of issuance: April 23, 2004.

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

Amendment No.: 129.

Facility Operating License No. NPF–58: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 1, 2003 (68 FR 15761).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 23, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 25, 2003, as supplemented by your letters dated June 16, 2003, January 14, February 23, and April 7, 2004.

Brief description of amendments: The amendments revise the technical specifications (TSs) to include implementation of relaxed axial offset control of the reactor core through changes in TS 3.2.1 and TS 3.2.3; relocation of selected operating parameters from TS 2.0, TS 3.1.8 and TS 3.3.1 to the Core Operating Limit Report (COLR) and the revised pressurizer pressure-low allowable value in TS Table 3.3.1–1. The TS changes also include, in TS 5.6.5, the topical reports documenting the Nuclear Regulatory Commission-approved methodologies that are used to support COLR implementation.

Date of issuance: April 28, 2004.

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

Amendment Nos.: 162 and 153.

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

Date of initial notice in ***Federal Register:*** April 29, 2003 (68 FR 22750).

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

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 28, 2004.

No significant hazards consideration comments received: No.

Pacific Gas and Electric Company, Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant, Units 1 and 2, San Luis Obispo County, California

Date of application for amendment: May 29, 2003, as supplemented by letters dated November 5, 2003 and December 23, 2003.

Brief description of amendments: The amendment revises Technical Specification 3.8.1, ''AC Sources-Operating,'' to extend the completion times for the required actions associated with restoration of an inoperable diesel generator (DG). Specifically, the changes extend the completion times for restoring an inoperable DG from 7 days to 14 days.

Date of issuance: April 20, 2004.

Effective date: April 20, 2004, and shall be implemented within 180 days of the date of issuance.

Amendment No.: Unit 1–166; Unit 2–167.

Facility Operating License Nos. DPR–80 and DPR–82: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** June 24, 2003 (68 FR 37581).

The supplemental letters dated November 5, 2003 and December 23, 2003, provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 20, 2004.

No significant hazards consideration comments received: No.

Pacific Gas and Electric Company, Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of application for amendments: February 28, 2003, as supplemented by letters dated October 30, 2003, December 2, 2003, and January 23, 2004.

Brief description of amendments: The amendments revise the Diablo Canyon Power Plant Technical Specifications (TS) to add a surveillance requirement to the Power Range Neutron Flux Rate—High Positive Rate Trip function.

Date of issuance: April 22, 2004.

Effective date: April 22, 2004, and shall be implemented within 180 days from the date of issuance.

Amendment Nos.: Unit 1—167; Unit 2—168.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 15, 2003 (68 FR 18283).

The October 30, 2003, December 2, 2003, and January 23, 2004, supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 22, 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: June 5, 2003.

Brief description of amendment: Revise the required actions in Technical Specification (TS) 3.6.1.9 when a containment purge or exhaust isolation valve is found inoperable as a result of leakage in excess of the limit. The changes allow alternate methods to ensure flow path isolation to the environment consistent with the methods allowed for containment isolation valves in TS 3.6.3, ''Containment Isolation Valves.''

Date of issuance: April 21, 2004.

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

Amendment Nos.: 290 & 280.

Facility Operating License No. DPR–77: Amendment revises the TSs.

Date of initial notice in ***Federal Register:*** July 8, 2003 (68 FR 40719).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 22, 2003, as supplemented March 19, 2004.

Brief description of amendment: The amendment revises Technical Specification 3.3.1, ''Reactor Trip System Instrumentation.'' The revision adds a Surveillance Requirement for response time to the Source Range Neutron Flux Reactor Trip function.

Date of issuance: April 19, 2004.

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

Amendment No.: 52.

Facility Operating License No. NPF–90: Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 18, 2003 (68 FR 54753). The supplemental letter provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 19, 2004.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 30th day of April 2004.

For the Nuclear Regulatory Commisison.

Ledyard B. Marsh,

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

[FR Doc. 04–10305 Filed 5–10–04; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF MANAGEMENT AND BUDGET

Revised Information Quality Bulletin on Peer Review

AGENCY: Office of Management and Budget, Executive Office of the President.

ACTION: Notice and request for comment: correction.

SUMMARY: This Notice provides the contact information and suggested approach for submitting comments on the ''Revised Information Quality Bulletin on Peer Review,'' published in the **Federal Register** on April 28, 2004 (69 FR 23230); this information was inadvertently omitted from the April 28th notice. As that notice indicated, interested parties should submit comments on or before May 28, 2004, to OMB's Office of Information and Regulatory Affairs. The April 28th notice contains the text of the proposed ''Revised Information Quality Bulletin on Peer Review'' as well as background and explanatory information.

ADDRESSES: Due to potential delays in OMB's receipt and processing of mail, respondents are strongly encouraged to submit comments electronically to ensure timely receipt. We cannot guarantee that comments mailed will be received before the comment closing date. Electronic comments may be submitted to: *OMB_peer_review@omb.eop.gov.* Please put the full body of your comments in the text of the electronic message and as an attachment. Please include your name, title, organization, postal address, telephone number, and e-mail address in the text of the message. Comments may also be submitted via facsimile to (202) 395–7245. Comments may be mailed to Dr. Margo Schwab, Office of Information and Regulatory Affairs, Office of Management and Budget, 725 17th Street, NW., New Executive Office Building, Room 10201, Washington, DC 20503.

FOR FURTHER INFORMATION CONTACT: Dr. Margo Schwab, Office of Information and Regulatory Affairs, Office of Management and Budget, 725 17th Street, NW., New Executive Office Building, Room 10201, Washington, DC 20503 (tel. (202) 395–3093).

John D. Graham,

Administrator, Office of Information and Regulatory Affairs.

[FR Doc. 04–10633 Filed 5–10–04; 8:45 am]

BILLING CODE 3110–01–P

SECURITIES AND EXCHANGE COMMISSION

Proposed Collection; Comment Request

Upon Written Request, Copies Available From: Securities and Exchange Commission, Office of Filings and Information Services, Washington, DC 20549.

Extension: Rule 6a–3, SEC File No. 270–0015, OMB Control No. 3235–0021.

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995,[1] the Securities and Exchange Commission (''Commission'') is soliciting comments on the collection of information summarized below. The Commission plans to submit this existing collection of information to the Office of Management and Budget for extension and approval.

Section 6 of the Exchange Act[2] sets out a framework for the registration and regulation of national securities exchanges. Under Commission Rule 6a–3,[3] one of the rules that implements Section 6, a national securities exchange (or an exchange exempted from registration as a national securities exchange based on limited trading volume) must provide certain supplemental information to the Commission, including any material (including notices, circulars, bulletins, lists, and periodicals) issued or made generally available to members of, or participants or subscribers to, the exchange. Rule 6a–3 also requires the exchanges to file monthly reports that set forth the volume and aggregate dollar amount of securities sold on the exchange each month.

The information required to be filed with the Commission pursuant to Rule 6a–3 is designed to enable the Commission to carry out its statutorily mandated oversight functions and to ensure that registered and exempt exchanges continue to be in compliance with the Act.

The respondents to the collection of information are national securities exchanges and exchanges that are exempt from registration based on limited trading volume.

The Commission estimates that each respondent makes approximately 25 such filings on an annual basis at an average cost of approximately $21 per response. Currently, 11 respondents (nine national securities exchanges and two exempt exchanges) are subject to the collection of information requirements of Rule 6a–3. The Commission estimates that the total burden for all respondents is 137.5 hours (25 filings/respondent per year × 0.5 hours/filing × 11 respondents) and $5775 ($21/response × 25 responses/respondent per year × 11 respondents) per year.

Written comments are invited on: (a) Whether the proposed collection of information is necessary for the proper performance of the functions of the agency, including whether the information shall have practical utility; (b) the accuracy of the agency's estimate of the burden of the proposed collection of information; (c) ways to enhance the

[1] 44 U.S.C. 3501 *et seq.*

[2] 15 U.S.C. 78f.

[3] 17 CFR 240.6a–3.