information: Michelle Schroll, 301–415–1662.

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

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

Dated: February 23, 2006.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 06–1908 Filed 2–24–06; 11:55 am]

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NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from February 3, 2006, to February 15, 2006. The last biweekly notice was published on February 14, 2006 (71 FR 7804).

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: Rulemaking and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and it is requested that copies be transmitted either by means of facsimile transmission to (301) 415–3725 or by email to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

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

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

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

Date of amendment request: November 30, 2005.

Description of amendment request: The proposed amendment would revise the frequency of the diesel generator automatic trips bypass surveillance requirement (SR) 3.8.1.11 from 18 months to 24 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, which is presented below:

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

No. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change decreases the frequency of SR 3.8.1.11, verification of the DG [diesel generator] automatic trips bypass, from 18 months to 24 months. The DG automatic trips bypass circuitry is required for DG operability and reliability during emergency operation of the DG. The proposed test frequency will continue to assure that the DG will perform as required. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated, because the factors that are used to determine the probability and consequences of accidents are not being affected.

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

No. The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated. There are no new or different accident initiators or sequences being created by the proposed Technical Specifications change. The required surveillance performed at the proposed frequency will continue to provide assurance that the trips bypass function is operable and is properly supporting operation of the associated DG. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The proposed change does not involve a significant reduction in the margin of safety. The proposed change will continue to ensure that the DG trips bypass function operates as designed. The functionality and operability of emergency power system is not being changed. 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 request involves no significant hazards consideration.

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

NRC Branch Chief: Michael L. Marshall, Jr.

Dominion Nuclear Connecticut, Inc., Docket No. 50–336, Millstone Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: January 4, 2006.

Description of amendment request: The proposed amendment would change the Millstone Power Station, Unit No. 2 Technical Specification (TS) 3/4.3.3.8, ''Instrumentation, Accident Monitoring,'' to modify the description of the pressurizer power operated relief valves (PORVs) and pressurizer safety valves position indicators.

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

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

Response: No.

The proposed amendment removes the wording ''Acoustic Monitor,'' which provides specific details related to system design, from items 4 and 6 of TS 3/4.3.3.8, Tables 3.3–11 and 4.3–7. The PORVs and Pressurizer Safety Valves position indicators (and the associated ''Acoustic Monitor'') provide only indications of valve position. They do not constitute a design feature that is an initial condition for a design basis accident or transient analysis. Furthermore, they do not affect the function of the system, equipment in the system or actuate to mitigate a design basis accident or transient. Therefore, the proposed changes do not increase the probability or consequences of an accident previously evaluated.

Additionally, the TS retains the requirement for the total and minimum channels required to be OPERABLE and to verify channel OPERABILITY at the designated frequencies. The PORVs and Pressurizer Safety Valves are equipped with positive position indication that meets the requirements of RG [Regulatory Guide] 1.97.

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

Response: No.

The proposed changes do not impact the capability of existing equipment to perform its intended functions. No system setpoints are being modified and no changes are being made to the method in which plant operations are conducted. No new failure modes that would impact accident analyses are introduced by the proposed changes. The proposed amendment does not introduce accident initiators or malfunctions that would cause a new or different kind of accident. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed amendment removes the wording ''Acoustic Monitor'' from items 4 and 6 of TS 3/4.3.3.8, Table[s] 3.3–11 and 4.3–7. The proposed changes do not affect any of the assumptions used in the accident analysis, nor does it affect any operability requirements for equipment important to plant safety. Therefore, the margin of safety is not impacted by the proposed amendment.

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

Attorney for licensee: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

NRC Branch Chief: Darrell J. Roberts.

Nebraska Public Power District, Docket No. 50–298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: December 30, 2005.

Description of amendment request: The proposed amendment establishes a combined leakage rate limit for the sum of the four Main Steam line leakage rates that is equal to four times the current individual Main Steam Isolation Valve (MSIV) leakage rate limit.

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

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

Response: No.

The proposed amendment does not involve a change to structures, systems, or components that would affect the probability of an accident previously evaluated in the Cooper Nuclear Station (CNS) Updated Safety Analysis Report (USAR). The proposed amendment results in no change in the radiological consequences of the design basis Loss-of-Coolant Accident (LOCA) as currently analyzed for CNS. That analysis was calculated for a combined Main Steam Isolation Valve (MSIV) leakage for determining acceptance to the regulatory limits for the offsite and Control Room radiation doses, as contained in 10 CFR 100 [Part 100 of Title 10 of the *Code of Federal Regulations*] and 10 CFR 50[,] Appendix A, General Design Criterion (GDC) 19. The aggregate Main Steam line leakage rate limit has no adverse effect on the environmental qualification of equipment important to safety, as provided for in 10 CFR 50.49.

Based on the above conclusions, this proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not modify the MSIVs or any other plant system or structure associated with this amendment and therefore, will not affect their capability to perform their design function. The combined total Main Steam line leakage rate is included in the current radiological analyses for the assessment of radiation exposure following an accident. This License Amendment Request revises the allowable leakage rate from a per valve limit to a total combined leakage rate limit for all four Main Steam lines but does not change the cumulative limit.

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

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

Response: No.

The leakage rate limit specified for the MSIVs is used to quantify the maximum amount of Secondary Containment bypass leakage assumed in the LOCA radiological analysis. Results of the analysis are evaluated against the dose limits contained in 10 CFR 50[,] Appendix A[,] GDC 19 and 10 CFR 100. The margin of safety in this context is considered to be the difference between the calculated dose exposures and the limits provided by GDC 19 and 10 CFR 100.

Therefore, since the proposed combined Main Steam line leakage rate limit is unchanged from the assumed maximum leakage rate for MSIVs, for the purpose of calculating [a] potential radiation dose, the margin of safety is not affected because the postulated radiation doses remain the same.

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

Attorney for licensee: Mr. John C. McClure, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Branch Chief: David Terao.

Nebraska Public Power District, Docket No. 50–298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: January 30, 2006.

Description of amendment request: The proposed change allows a delay time for entering a supported system Technical Specification (TS) when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed consistent with the program in place for complying with the requirements of 10 CFR 50.65(a)(4). Limiting Condition for Operation (LCO) 3.0.8 is added to the TS to provide this allowance and define the requirements and limitations for its use.

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–372, Revision 4. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on November 24, 2004 (69 FR 68412), on possible amendments concerning TSTF–372, 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 May 4, 2005 (70 FR 23252). The licensee affirmed the applicability of the following NSHC determination in its application dated January 30, 2006.

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 a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. Therefore, the probability of an accident previously evaluated is not significantly increased, if at all. The consequences of an accident while relying on allowance provided by proposed LCO 3.0.8 are no different than the consequences of an accident while relying on the TS required actions in effect without the allowance provided by proposed LCO 3.0.8. 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). Allowing delay times for entering supported system TS when inoperability is due solely to inoperable snubbers, if risk is assessed and managed, 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 a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in Regulatory Guide 1.177. A bounding risk assessment was performed to justify the proposed TS changes. The proposed LCO 3.0.8 defines limitations on the use of the provision and includes a requirement for the licensee to assess and manage the risk associated with operation with an inoperable snubber. 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: Mr. John C. McClure, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Branch Chief: David Terao.

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

Date of amendment request: November 11, 2005.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3.6.5, ''Containment Spray and Cooling Systems''; an existing Condition, two Surveillance Requirements, and add a new Condition which will allow continued plant operation with TS limitations when two Containment Cooling System fan coil units (FCUs), one in each train, are inoperable.

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

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

Response: No.

This license amendment proposes to revise the Technical Specifications to allow plant operation to continue for a limited time period under Technical Specification controls with two fan coil units, one fan coil unit from each containment cooling train, providing the required cooling function. Analyses demonstrate that any two fan coil units, whether they are in the same train or from opposite trains, are sufficient to supply the required containment cooling following a design basis accident when the plant in the proper configuration as required by the proposed Technical Specifications.

The containment cooling system is required for accident mitigation and is not an accident initiator, thus revising the equipment required to provide the safety function does not involve a significant increase in the probability of an accident previously evaluated.

Since the proposed change continues to provide the post-accident containment cooling function under Technical Specification controls, this change does not involve an increase in the consequences of an accident. Thus this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

This license amendment proposes to revise the Technical Specifications to allow plant operation to continue for a limited time period under Technical Specification controls with two fan coil units, one fan coil unit from each containment cooling train, providing the required cooling function. Analyses demonstrate that any two fan coil units, whether they are in the same train or from opposite trains, are sufficient to supply the required containment cooling following a design basis accident when the plant in the proper configuration as required by the proposed Technical Specifications.

The proposed licensing basis changes do not involve a change in the function or use of the containment cooling system. It does assure that the containment cooling function is provided during plant operations for post-accident mitigation. There are no new failure modes or mechanisms created through allowing different combinations of fan coil units to provide the cooling function as proposed by this Technical Specification change. There are no new accident precursors generated by providing the required cooling function with an operable fan coil unit from each train.

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

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

Response: No.

This license amendment proposes to revise the Technical Specifications to allow plant operation to continue for a limited time period under Technical Specification controls with two fan coil units, one fan coil unit from each containment cooling train, providing the required cooling function. Analyses demonstrate that any two fan coil units, whether they are in the same train or from opposite trains, are sufficient to supply the required containment cooling following a design basis accident when the plant in the proper configuration as required by the proposed Technical Specifications.

Current plant Technical Specifications allow plant operation to continue for 7 days with the containment cooling function provided by the two operable fan coil units of a single operable containment cooling train. This is acceptable because engineering analyses demonstrate that the two fan coil units of a single train can provide the required post-accident containment cooling.

Likewise, engineering analyses demonstrate that any two fan coil units from opposite containment cooling trains can also provide the required post-accident containment cooling if the cooling water flow to the other fan coil unit in each train is isolated. This license amendment request proposes Technical Specifications which will allow plant operation to continue for 7 days with the containment cooling function provided by two fan coils from opposite trains provided the cooling water flow to the other fan coil unit in each train is isolated. Thus, from a cooling capacity perspective, this proposed Technical Specification change does not involve a reduction in a margin of safety.

When inoperable plant systems are under Technical Specification controls that limit the time for inoperability, a single failure in addition to the inoperable equipment is not postulated. Therefore, whether two inoperable fan coil units are in the same train or opposite trains does not change the availability of the two remaining operable fan coil units. Thus from a Technical Specification perspective, this proposed Technical Specification change does not involve a reduction in a margin of safety.

Therefore, based on the considerations given above, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Acting Branch Chief: Timothy J. Kobetz.

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

Date of amendment request: December 19, 2005.

Description of amendment request: The proposed change will revise Fort Calhoun Station, (FCS) Technical Specification 2.4, ''Containment Cooling,'' (and associated Bases) to reduce the required number of operable Containment Spray (CS) pumps from three to two in order to enhance net positive suction head (NPSH) margins. This change will be accomplished by disabling the containment spray actuation signal (CSAS) automatic start feature of CS pump SI–3C. This change will reduce the head loss across the containment sump strainers during the recirculation phase of a design-basis accident (DBA) by reducing flow rates, and will improve NPSH available ($NPSH_A$).

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

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

Response: No.

The Containment Spray (CS) system is not an initiator of any accident previously evaluated at the Fort Calhoun Station (FCS); the CS system is an accident mitigation system. The CS system's licensing basis functions are to limit the containment pressure rise and reduce the leakage of airborne radioactivity from the containment by providing a means for cooling the containment following a loss-of-coolant accident (LOCA) or main steam line break (MSLB) inside containment. The proposed change disables the CSAS automatic start feature of one of the three CS pumps.

The only FCS safety analysis that currently assumes three CS pumps operating to mitigate an accident is the Containment Pressure Analysis for a[n] MSLB inside containment. Even though this analysis assumes operation of all three CS pumps, it also shows that peak containment pressure occurs prior to the CS system starting, therefore, the CS system does not mitigate the peak pressure for a[n] MSLB. The reviews evaluated both existing AORs [analyses of record] and those analyses developed for the Steam Generator Replacement (RSG) project. The analysis developed for the RSG project that evaluates the Containment Pressure Analysis for MSLB inside containment was reviewed for the impact of reducing the number of operating CS pumps from three to two. This review determined that the RSG MSLB analysis will be acceptable and will continue to be bounded by the analysis currently documented in USAR. AOR peak pressure is unaffected by implementation of this proposed change. Therefore, the combination of the RSG project and this containment spray modification will not result in an increase in the currently documented peak containment pressure for an MSLB. Therefore, the evaluation for the MSLB event has determined that the containment pressure response is acceptable with less than three CS pumps operating.

The LOCA analysis source term is based on operation of minimum safeguards due to a worst-case single failure. The minimum safeguards configuration is unchanged by this modification. Following implementation of the proposed change at least one CS pump will be available to mitigate a LOCA as currently assumed in the analysis, therefore, the proposed change will have no adverse effect on the radiological consequences following a LOCA. The analyses that establish the radiological consequences for the site are based on a Large Break LOCA with a single CS pump in operation, therefore, single CS pump operation during a[n] MSLB inside containment is bounded by the LOCA analysis.

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

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

Response: No.

The proposed change will reduce the number of operable CS pumps from three to two; however, previous accident analyses will remain valid. No credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing basis have been created and none of the initial condition assumptions of any accident evaluated in the safety analysis are impacted.

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

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

Response: No.

The containment building and associated penetrations are designed to withstand an internal pressure of 60 psig at 305 °F, including all thermal loads resulting from the temperature associated with this pressure, with a leakage rate of 0.1 percent by weight or less of the contained volume per 24 hours. The CS System and the Containment Fan Coolers are credited for maintaining containment pressure and temperatures within design limitations, and assure that the release of fission products to the environment following a design[-]basis accident will not exceed regulatory guidelines. The FCS licensing basis credits only one of the three CS pumps to limit the containment pressure to below the design value for a LOCA. Currently, the FCS licensing basis credits three CS pumps for a[n] MSLB, however, the CS system is not credited for limiting peak containment pressure for a[n] MSLB.

The EEQ [electrical equipment qualification] profile developed for the current plant configuration bounds those associated with the upcoming RSG modification. Both the proposed CS system changes and the RSG projects are scheduled for the same refueling outage. The thermal lag analysis of equipment performed using the current plant configuration demonstrated a large margin between the equipment evaluated during the accident versus the conditions under which it was tested. The RSG modification will further increase this margin. As part of the RSG effort the EEQ analysis will be revised to address RSG issues and will include the changes to containment spray. When the margins associated with the current analysis as well as increases in margin when the new analysis is implemented it is expected that the changes to the containment spray system will not produce an adverse result. All equipment will remain qualified to operate in the accident environment.

Additionally, the CFCs [containment fan coolers] operate independently of the CS system to remove heat from the containment atmosphere. The CFCs consist of two redundant trains; each train with one air cooling and filtering unit and one air cooling unit, for a total of four cooling units. Operation of the CFCs is credited in the MSLB containment pressure analysis. The CFCs are not impacted by this proposed change. During the MSLB containment spray takes place after the peak containment pressure occurs. Therefore, the licensing basis capabilities of the Containment Cooling System, which consists of the CS and CFCs, is not adversely affected by the proposed change; the ability to maintain containment peak pressure and temperature and long[-]term containment pressure and temperature will be maintained.

Particulate fission products that are released into the containment following a DBA are removed by the CS system for those events that result in CS actuation. The water spray strips radioactive particles from the atmosphere where they fall to the floor and are washed into the containment sump. The radiological consequences analysis credits CS system operation for removal of particulates from the containment atmosphere during a LOCA. The LOCA analysis source term is based on operation of minimum safeguards due to a worst-case single failure, and a presumption of core damage. Minimum safeguards corresponds to one CS pump and one CS header operation and take into account pump degradation, and instrument uncertainties. The analyses that establish the radiological consequences for the site are not impacted by the proposed modification. These analyses are based on a Large Break LOCA with a single CS pump in operation. Therefore, single CS pump operation bounds the plant configuration following the proposed modification.

The Large Break LOCA assumes that there will be three CS pumps operating when evaluating the effects of containment pressure on ECCS [emergency core cooling system] performance. The analysis assumes three CS pumps, which minimizes containment pressure, to conservatively evaluate ECCS performance in response to a LOCA. The use of two CS pumps versus three improves ECCS performance and thus increases margin to 10 CFR 50.46 limits on peak clad temperature.

In summary, following implementation of the proposed change:

- Peak containment pressure for analyzed DBAs will not be increased;
- The assumptions used in the environmental qualification of equipment exposed to the containment atmosphere following a DBA remaining bounding; and
- The radiological consequences for the bounding DBA remains unchanged.
- The currently calculated peak clad temperature following a LOCA remains bounded by existing analysis.

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

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

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

NRC Branch Chief: David Terao.

Pacific Gas and Electric Company, Docket No. 50–323, Diablo Canyon Nuclear Power Plant, Unit No. 2, San Luis Obispo County, California

Date of amendment requests: January 13, 2006.

Description of amendment requests: The proposed amendment would revise Technical Specification 5.6.5, ''Core Operating Limits Report (COLR),'' by adding WCAP–16009–P–A, ''Realistic Large-Break LOCA [Loss-of-Coolant Accident] Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM),'' dated January 2005, as an approved analytical method for determining core operating limits for Unit 2.

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

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

Response: No.

The proposed change to allow the use of the best estimate loss-of-coolant accident (LOCA) analysis methodology using the automated statistical treatment of uncertainty methodology (ASTRUM) does not involve a physical alteration of any plant equipment or change operating practice at Unit 2 of Diablo Canyon Power Plant (DCPP). Therefore, there will be no increase in the probability of a LOCA. The consequences of a LOCA are not being increased.

The plant conditions assumed in the analysis are bounded by the design conditions for all equipment in Unit 2. That is, it is shown that the emergency core cooling system is designed so that its calculated cooling performance conforms to the criteria contained in 10 CFR [Title 10 of the Code of Federal Regulations, Section] 50.46, paragraph b. No other accident is potentially affected by this change.

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 accident from any accident previously evaluated?

Response: No.

The proposed change would not result in any physical alteration to any Unit 2 system, and there would not be a change in the method by which any safety [-]related system performs its function. Analyses of transient events have confirmed that no transient event results in a new sequence of events that could lead to a new accident scenario. The parameters assumed in the analysis are within the design limits of existing plant equipment.

In addition, employing the ASTRUM methodology does not create any new failure modes that could lead to a different kind of accident. The design of all systems remains unchanged and no changes are being made to any reactor protection system or engineered safeguard features actuation setpoints.

Based on this review, it is concluded that no new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes.

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

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

Response: No.

It has been shown that the analytic technique used in the analysis realistically describes the expected behavior of the DCPP Unit 2 reactor system during a postulated LOCA. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of LOCAs with different break sizes, different locations, and other variations in properties have been analyzed to provide assurance that the most severe postulated LOCAs were analyzed. The analysis has demonstrated that all acceptance criteria contained in 10 CFR 50.46[,] paragraph b continue to be satisfied.

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 requests involve no significant hazards consideration.

Attorney for licensee: Richard F. Locke, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Branch Chief: David Terao.

Pacific Gas and Electric Co., Docket No. 50–133, Humboldt Bay Power Plant (HBPP), Unit 3 Humboldt County, California

Date of amendment request: January 19, 2006.

Description of amendment request: The licensee has proposed to revise the Technical Specifications (TS) to correct an editorial error in TS 3.1.2, ''Spent Fuel Pool Load Restrictions,'' and to change TS 5.2.2, ''Facility Staff,'' to allow the Unit 3 control room to be temporarily unmanned during emergency conditions that require personnel to evacuate buildings for their safety.

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

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

Response: No.

The proposed editorial change has no impact on probability or consequences of accidents. The following discussion applies to the proposed change related to control room evacuation.

Allowing plant personnel to not continuously man the control room has no impact on the probability of an accident from occurring, especially acts of nature such as earthquakes and tsunamis.

The HBPP DSAR, Appendix A, and NRC SER, Section 10, dated April 29, 1987, evaluate various accidents at HBPP. Because all fuel has been removed from the reactor vessel and stored in the spent fuel pool, the majority of accidents analyzed pertain to events that could only affect spent fuel or the spent fuel pool. All accidents affecting spent fuel or the spent fuel pool do not require operator action to protect the public health and safety or to maintain offsite radiological doses well within regulatory limits. In addition, NRC SER, Section 10.7, ''Impact of Tsunami Flooding,'' analyzes the impact of tsunami flooding. That analysis identifies a likely impact of the tsunami to be a release of the radwaste tank radionuclide contents to the bay and some damage to the reactor building. For both situations, no operator action is required to maintain offsite radiological doses well within regulatory limits.

Allowing the control room to be temporarily unmanned under emergency conditions does not create problems that could increase the consequences of an accident. The primary function of manning the control room is for an operator to observe and acknowledge alarms. Recovery actions to respond to damage to spent fuel, the spent fuel pool, or radwaste tanks are taken by personnel outside the control room. No recovery actions are required to be taken by the control room operator to respond to damage to spent fuel, the spent fuel pool, or radwaste tanks.

Evacuating occupied buildings, including the control room, during a tsunami, allows the control room operator to return to the control room after the tsunami and assess damage by observing indicators and alarms. Upon returning to the control room, the operator would be able to direct and monitor recovery efforts from the control room that may be necessary to bring plant parameters within required specifications.

If an operator remains in the control room during a tsunami and becomes injured, that operator would be unable to direct and monitor recovery efforts. Under this scenario, other plant personnel who evacuated to higher ground onsite within the OCA would eventually return to the plant, including the control room, and perform any required recovery functions. Therefore, consequences of a tsunami are not increased by not continually manning the control room during the event.

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

Response: No.

The proposed editorial change has no impact on accidents. The following discussion applies to the proposed change related to control room evacuation.

As discussed in the response to question 1 above, none of the analyzed accidents require operator action to keep offsite radiological doses well within regulatory limits. In addition, allowing plant personnel to not continuously man the control room after an emergency situation has occurred, has no impact on the possibility of a new or different kind of accident from occurring. If the plant is evacuated, no work activities will be performed in the plant. With the plant in SAFSTOR and no work being performed, there are no actions required to be taken by personnel manning the control room.

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

Response: The proposed editorial change has no impact on margin of safety. The following discussion applies to the proposed change related to control room evacuation.

NRC SER Section 10.8, ''Accident Analysis Conclusions,'' summarizes the consequences from accidents in terms of offsite radiological doses. SER Section 10.8 includes the statement, ''The (NRC) staff has determined that offsite radiological consequences due to a tsunami are within acceptable dose guideline values.'' As discussed in the response to question 1 above, none of the analyzed accidents require operator action to keep offsite radiological doses well within regulatory limits. Therefore, temporarily not manning the control room during an emergency will have no impact on the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based upon the staff's review of the licensee's analyses as well as the staff's own evaluation, the staff concludes 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: Richard F. Locke, Esquire, Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Section Chief: Claudia Craig.

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

Date of amendment request: January 31, 2006.

Description of amendment request: The amendments would revise Technical Specification (TS) 3.8.3.1, ''Onsite Power Distribution-Operating,'' to extend the allowed outage time (AOT) for an inoperable Class 1E vital 120-volt alternating current inverter. The TS currently provides an AOT of 24 hours to restore an inoperable inverter. Based on risk-informed assessment, the amendments would extend the AOT to 7 days.

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 formatting changes to TS 3.8.3.1 Action b and the change to the AOT for an inoperable inverter to be extended from 24 hours to 7 days do not alter any plant equipment or operating practices in such a manner that the probability of an accident is increased. The proposed changes will not alter assumptions relative to the mitigation of an accident or transient event.

An evaluation was performed to determine the risk significance of the proposed change to the AOT. The risk evaluation concludes that the ΔCDF [core damage frequency] and ΔLERF [large early release frequency] associated with the proposed changes are 1.88E–07 and 2.05E–09, respectively, which are characterized as ''very small changes'' by RG [Regulatory Guide] 1.174. The ICCDP [incremental conditional core damage probability] and ICLERP [incremental conditional large early release probability] associated with the proposed change are 3.63E–07 and 1.08E–08, respectively, which are within the acceptance criteria in RG 1.177. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, reactor coolant pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed change to TS 3.8.3.1 to allow the AOT for an inoperable inverter to be extended from 24 hours to 7 days has been evaluated for its effect on plant safety. The risk-informed evaluation concludes that the ΔCDF and ΔLERF associated with the proposed change are 1.88E–07 and 2.05E–09, respectively, which are characterized as ''very small changes'' by RG 1.174. The ICCDP and ICLERP associated with the proposed change are 3.63E–07 and 1.08E–08, respectively, which are within the acceptance criteria in RG 1.177. The proposed changes to the formatting of TS 3.8.3.1 Action b are administrative only and have no impact on margin of safety. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

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

NRC Branch Chief: David Terao.

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

Date of amendment request: December 14, 2005 (TS–05–07).

Description of amendment request: The proposed amendment would revise Technical Specification Section 5.7.2.19, ''Containment Leakage Rate Testing Program,'' to allow a one time, 5-year extension to the current 10-year test interval for the performance-based leakage rate test program for 10 CFR Part 50, Appendix J, Type A tests.

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

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

Response: No.

The proposed change for extending Type A test frequency does not significantly increase the probability of an accident previously evaluated since the change is not a modification to plant systems, nor a change to plant operation that could initiate an accident.

TVA performed an evaluation of the risk significance for the proposed increase to the WBN Unit 1 Type A test frequency. The results of the TVA risk evaluation indicates that the increase in Large Early Release Frequency (LERF) remains below the level of risk significance defined in the NRC Regulatory Guide 1.174, ''An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis.'' TVA's evaluation indicates that the calculated increase in frequency for all releases (small, large, early and late) and the increase in radiation dose to the population are also non-risk significant.

The proposed test interval extension does not involve a significant increase in the consequences of an accident. Research documented in NUREG–1493, ''Performance-Based Containment Leakage-Test Program,'' determined that generically, very few potential containment leakage paths fail to be identified by Type A tests. An analysis of 144 Type A test results, including 23 failures, found that no failures were due to containment liner breach. The NUREG concluded that reducing the Type A test frequency to once per 20 years would lead to an imperceptible increase in risk. Furthermore, the NUREG concluded that Type B and C testing provides assurance that containment leakage from penetration leak paths (*i.e.*, valves, flanges, containment air-locks) identify any leakage that would otherwise be detected by the Type A tests.

In addition to the NUREG conclusions, TVA's American Society of Mechanical Engineers (ASME) IWE program performs containment inspections in order to detect evidence of degradation that may either affect the containment structural integrity or leak tightness.

Therefore, the proposed extension of the Type A test interval does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change to extend the Type A test interval does not create the possibility of a new or different type of accident because there are no physical changes made to the plant or plant equipment governing normal plant operation. There are no changes to the operation of the plant that would introduce a new failure mode creating the possibility of a new or different kind of accident. Therefore, the proposed extension 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 change to extend the Type A test interval will not significantly reduce the margin of safety. A generic study documented in NUREG–1493 indicates that extending the Type A leak test interval to 20 years would result in an imperceptible increase in risk to the public. The NUREG also found that, generically, the containment leakage rate contributes a very small amount to the individual risk and that the decrease in the Type A test frequency would have a minimal effect on risk because most potential leakage paths are detected by Type C testing.

Previous Type A leakage tests conducted on WBN Unit 1 indicate that leakage from containment have been less than the 10 CFR 50, Appendix J leakage limit of 1.0 L_a. A review of the previous Type A test results indicate a stable trend with an increase of less than 15 percent of L_a, well below the 1.0 L_a leakage limit.

Therefore, these test results, in conjunction with the research findings from NUREG–1493, provide assurance that the proposed extension to the Type A test interval 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: Michael L. Marshall, Jr.

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

Date of application request: August 26, 2005, as supplemented by letter dated December 16, 2005.

Description of amendment request: The amendment would authorize changes to the Final Safety Analysis Report (FSAR) for the Callaway Plant, Unit 1, that would revise the methodology for the reactor coolant system (RCS) leak detection instrumentation. This revision would clarify the requirements of the containment atmosphere gaseous radioactivity monitor with regard to the RCS leak detection capability and would justify that the monitor can be considered operable in compliance with Limiting Condition for Operation 3.4.15, in Technical Specification (TS) 3.4.15, ''RCS Leakage Detection Instrumentation,'' during all applicable reactor modes. There are no proposed changes 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. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change has been evaluated and determined to not increase the probability or consequences of an accident previously evaluated. The proposed change does not make hardware changes and does not alter the configuration of any plant system, structure, or component (SSC). The proposed change only clarifies the design and OPERABILITY requirements for the containment atmosphere gaseous radioactivity monitor[s] and identifies the capabilities of the containment atmosphere gaseous radioactivity monitors at low RCS [radio]activity levels. The containment radiation monitors are not initiators of any accident; therefore, the probability of occurrence of an accident is not increased. The FSAR and TS will continue to require diverse means of [RCS] leakage detection equipment, thus ensuring that leakage due to cracks [in the RCS] would continue to be identified prior to propagating to the point of a [RCS] pipe break. Therefore, the consequences of an accident [previously evaluated] are not increased.

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

The proposed change does not involve the use or installation of new equipment and the currently installed equipment will not be operated in a new or different manner. No new or different system interactions are created and no new processes are introduced. The proposed changes will not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bas[i]s [for the Callaway Plant]. The proposed change does not affect any SSC associated with an accident initiator. Based on this evaluation, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not alter any RCS leakage detection components. The proposed change only clarifies the design and OPERABILITY requirements for the containment atmosphere gaseous radioactivity monitor[s] and identifies the capabilities of the containment atmosphere gaseous radioactivity monitors at low RCS [radio]activity levels. This change is required since the level of radioactivity in the Callaway Plant reactor coolant has become much lower than what was assumed in the FSAR [when the plant was licensed] and the gaseous channel [(monitor)] can no longer promptly detect a small RCS leak under all operating conditions. The proposed amendment continues to require diverse means of [RCS] leakage detection equipment with [the] capability to promptly detect RCS leakage. Although not required by TS, additional diverse means of leakage detection capability are available as described in the FSAR Section 5.2.5. Early detection of [RCS] leakage, as the potential indicator of a crack(s) in the RCS pressure boundary, will thus continue to be in place so that such a condition is known and appropriate actions taken well before any such crack would propagate to a more severe condition. Based on this evaluation, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the 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 Branch Chief: David Terao.

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

Date of amendment request: February 1, 2006.

Description of amendment request: The amendment would revise the Inservice Testing Program in Section 5.5.8 of the Administrative Controls, Programs and Manuals, section of the Technical Specifications (TSs). The licensee is adopting NRC-approved Technical Specification Task Force (TSTF) 479, Revision 0, ''Changes to Reflect Revision of 10 CFR 50.55a.''

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

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

Response: No.

The proposed change revises TS 5.5.8, ''Inservice Testing Program,'' for consistency with the requirements of 10 CFR 50.55a(f) regarding the inservice testing of pumps and valves. The proposed change incorporates revisions to the ASME [American Society of Mechanical Engineers] Code [for Operation and Maintenance of Nuclear Power Plants] that result in a net improvement in the measures for testing pumps and valves.

The proposed change does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. They do not involve the addition or removal of any equipment, or any design changes to the facility. Therefore, the proposed change does not represent a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change revises TS 5.5.8, ''Inservice Testing Program,'' for consistency with the requirements of 10 CFR 50.55a(f) regarding the inservice testing of pumps and valves. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves.

The proposed change does not involve a modification to the physical configuration of the plant (*i.e.*, no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure. Therefore, this proposed change does not create the possibility of an accident of a different kind than previously evaluated.

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

Response: No.

The proposed change revises TS 5.5.8, ''Inservice Testing Program,'' for consistency with the requirements of 10 CFR 50.55a(f) regarding the inservice testing of pumps and valves. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves. The safety function of the affected pumps and valves will be maintained. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: David Terao.

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

Date of amendment request: February 7, 2006.

Description of amendment request: The amendment would add Surveillance Requirement (SR) 3.3.1.16, to verify the reactor trip system response time, to Function 3.a, power range neutron flux—high positive rate trip function, in Table 3.3.1–1, ''Reactor Trip System Instrumentation,'' of the Technical Specifications (TSs).

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.

Overall protection system performance will remain within the bounds of the accident analysis since there are no hardware changes. The design of the Reactor Trip System (RTS) instrumentation, specifically the positive [neutron] flux rate trip (PFRT) function, will be unaffected. The reactor protection system will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request [(*i.e.*, this amendment application)] are maintained.

The proposed change imposes additional surveillance requirements to assure safety related structures, systems, and components are verified to be consistent with the [plant] safety analysis and licensing basis. In this specific case, a response time verification requirement will be added to the PFRT Function [in TS Table 3.3.1–1].

The proposed [change] will not modify any system interface. The proposed [change] will not affect the probability of any event initiators. There will be no degradation in the performance of or an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance. The proposed [change] will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the Updated Safety Analysis Report (USAR) [for Wolf Creek Generating Station].

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 or maintained. The proposed [change does] not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed [change does] not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed [change is] consistent with 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.

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

Response: No.

There are no hardware changes nor are there any changes in the method by which any safety related plant system performs its safety function. This change will not affect the normal method of plant operation or change any operating parameters. No performance requirements will be affected; however, the proposed change does impose additional surveillance requirements. The additional requirements are consistent with assumptions made in the safety analysis and licensing basis.

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

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

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

Response: No.

The proposed [change does] 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, DNBR [departure from nucleate boiling ratio] limit, F_Q [heat flux hot channel factor], $F\Delta H$ [nuclear enthalpy rise hot channel factor], LOCA PCT [loss-of-coolant accident peak cladding temperature], peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the [NRC] Standard Review Plan [NUREG–0800] will continue to be met.

The safety analysis limits assumed in the transient and accident analyses are unchanged. None of the acceptance criteria for any accident analysis is changed. The imposition of additional surveillance requirements increases the margin of safety by assuring that the affected safety analysis assumptions on equipment response time are verified on a periodic frequency. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: David Terao.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) The applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland.

Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of application for amendments: August 11, 2005.

Brief Description of amendments: The amendments revise Technical Specification (TS) 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' by removing an exception that allows for compensation of flow meter instrument inaccuracies in accordance with ANSI/ANS–56.8–1987 rather than ANSI/ANS–56.8–1994.

Date of issuance: February 8, 2006.

Effective date: Date of issuance to be implemented within 60 days.

Amendment Nos.: 238 and 266.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments change the TS.

*Date of initial notice in **Federal Register:*** September 13, 2005 (70 FR 54087).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 8, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 8, 2005.

Brief description of amendment: The proposed changes would add Limiting Condition for Operation 3.0.8 to address conditions where one or more snubbers are unable to perform their associated support function.

Date of issuance: February 13, 2006.

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

Amendment Nos.: 245 and 229.

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

*Date of initial notice in **Federal Register:*** August 16, 2005 (70 FR 48203).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 13, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 7, 2005, as supplemented by letter dated December 5, 2005.

Brief description of amendments: The amendments will add two Nuclear Regulatory Commission (NRC) approved topical report references to the list of analytical methods in Technical Specification 5.6.5, ''Core Operating Limits Report,'' that can be used to determine core operating limits.

Date of issuance: February 1, 2006.

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

Amendment Nos.: 174 and 160.

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 16, 2005 (70 FR 48205).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 1, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 17, 2004.

Brief description of amendments: The amendments revised the Appendix B, Environmental Protection Plan (non-radiological), of the Quad Cities Station Renewed Facility Operating Licenses.

Date of issuance: February 2, 2006.

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

Amendment Nos.: 229 and 224.

Facility Operating License Nos. DPR–29 and DPR–30: The amendments revised the Environmental Protection Plan.

*Date of initial notice in **Federal Register:*** April 12, 2005 (70 FR 19115).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 2, 2006.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1 (BVPS–1), Beaver County, Pennsylvania

Date of application for amendment: April 13, 2005, as supplemented by letters dated August 26, October 28 and 31, November 18, and December 6 and 16, 2005.

Brief description of amendment: The amendment revised the Technical Specifications (TSs) to allow replacement of the BVPS–1 steam generators (SGs). These changes include revising the fuel assembly-specific departure from nucleate boiling ratios and correlations, modifying the Overtemperature ΔT and Overpower ΔT equations, revising the SG water level low-low and high-high setpoints, revising the SG secondary side level in Modes 4 and 5, revising the SG TSs to reflect the replacement SGs and remove TS requirements that are no longer applicable to the new SGs, revising the required charging pump discharge pressure for reactor coolant pump seal injection flow, raising the accumulator pressure, and adding WCAP–14565–P–A (VIPRE) and WCAP–15025–P–A (WRB–2M) Topical Reports to the list of NRC-approved methodologies listed in TS 6.9.5. The amendment also approves an expanded selective alternate source term methodology implementation in accordance with Regulatory Guide 1.183, ''Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,'' and approves use of the 1979 ANS Decay Heat + 2σ model for mass and energy releases for a main steam line break outside containment.

Date of issuance: February 9, 2005.

Effective date: As of its date of issuance and shall be implemented prior to entry into Mode 4 upon startup from refueling outage 1R17 which begins on or about February 10, 2006.

Amendment No: 273.

Facility Operating License No. DPR–66: The Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** June 21, 2005 (70 FR 35737). The supplements dated August 26, October 28 and 31, November 18, and December 6 and 16, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 9, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 4, 2004, as supplemented July 8, and November 14, 2005.

Brief description of amendments: These amendments approved application of the Westinghouse best-estimate loss-of-coolant accident (LOCA) analysis methodology to BVPS–1 and 2 for large-break LOCA analysis.

Date of issuance: February 6, 2006.

Effective date: These license amendments are effective as of the date of issuance and shall be implemented for BVPS–1, prior to Mode 4 entry during startup from refueling outage 1R17 which begins on or about February 10, 2006, and for BVPS–2, prior to Mode 4 entry during startup from refueling outage 2R12 which begins October 2006.

Amendment Nos.: 272 and 154.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 7, 2004 (69 FR 70718). The supplements dated July 8, and November 14, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 6, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 10, 2005.

Brief description of amendment: The amendments deleted the power range neutron flux high negative rate trip function from Table 3.3.1–1, ''Reactor Trip System Instrumentation.''

Date of issuance: February 10, 2006.

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

Amendment No.: 293, 275.

Facility Operating License No. DPR–58: Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 6, 2005 (70 FR 72674). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 10, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 7, 2004, as supplemented by letters dated February 18, May 20, June 16, July 8, August 3, September 23, and November 16, 2005, and February 6, 2006.

Brief description of amendment: The amendment revised the Technical Specifications (TSs) to reflect an expanded operating domain resulting from the implementation of the Average Power Range Monitor, Rod Block Monitor TSs/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA).

Date of issuance: February 8, 2006.

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

Amendment No.: 163.

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

*Date of initial notice in **Federal Register:*** September 14, 2004 (69 FR 55471). The supplements dated February 18, May 20, June 16, July 8, August 3, September 23, and November 16, 2005, and February 6, 2006, provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 8, 2006.

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: February 1, 2005, supplemented by letters dated February 22, September 16, December 2, 2005, and January 5, 2006.

Brief description of amendments: The amendments revise the spent fuel pool (SFP) criticality analysis methodology and technical specifications governing the storage of irradiated fuel in the SFP. The licensee's amendment request stated that subcritical conditions would be maintained in the SFP under the revised technical specification storage requirements.

Date of issuance: February 5, 2006.

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

Amendment Nos.: 172, 162.

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

*Date of initial notice in **Federal Register:*** March 15, 2005, (70 FR 12748). 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 February 5, 2006.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendments request: January 19, 2005, as supplemented on June 9 (two letters) and November 18, 2005.

Brief Description of amendments: The amendment authorizes revision of the Updated Final Safety Analysis Report (UFSAR) to reflect the utilization of fire-rated electrical Mineral Insulated cables in lieu of Appendix R, Section III.G.2 1-hour rated fire barriers.

Date of issuance: February 13, 2006.

Effective date: As of the date of issuance, to be incorporated into the UFSAR at the time of its next update.

Amendment No.: 162.

Renewed Facility Operating License Nos. NPF–2 and NPF–8: Amendment authorizes revision to the UFSAR.

*Date of initial notice in **Federal Register:*** April 26, 2005 (70 FR 21464). The supplemental letters 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 amendments is contained in a Safety Evaluation dated February 13, 2006.

No significant hazards consideration comments received: No.

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

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 for the amendment 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.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective 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 application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management 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 PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov.*

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. 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, and electronically on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If there are problems in accessing the document, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737, or by e-mail to *pdr@nrc.gov.* If a request for a hearing or petition for leave to intervene is filed by the above date, 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 identify 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 intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner 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.[1] 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 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.

Each contention shall be given a separate numeric or alpha designation within one of the following groups:

1. *Technical*—primarily concerns/issues relating to technical and/or health and safety matters discussed or referenced in the applications.
2. *Environmental*—primarily concerns/issues relating to matters discussed or referenced in the environmental analysis for the applications.
3. *Miscellaneous*—does not fall into one of the categories outlined above.

As specified in 10 CFR 2.309, if two or more petitioners/requestors seek to co-sponsor a contention, the petitioners/requestors shall jointly designate a representative who shall have the authority to act for the petitioners/requestors with respect to that contention. If a petitioner/requestor seeks to adopt the contention of another sponsoring petitioner/requestor, the petitioner/requestor who seeks to adopt the contention must either agree that the sponsoring petitioner/requestor shall act as the representative with respect to that contention, or jointly designate with the sponsoring petitioner/requestor a representative who shall have the authority to act for the petitioners/requestors with respect to that contention.

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. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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 or 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).

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

Date of amendment request: February 5, 2006, as supplemented February 5, 2006.

Description of amendment request: The amendment revised Technical Specification 3.8.1, ''AC Sources—Operating,'' to extend the allowed outage time for Emergency Diesel Generator 12 from seven days to 14 days for one specific incident.

Date of issuance: February 6, 2006.

Effective date: As of the date of issuance and shall be implemented immediately.

Amendment No.: 171.

Facility Operating License No. 50–341: Amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): No. The Commission's related evaluation of the amendment, finding of emergency circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated February 6, 2006.

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

NRC Branch Chief: Timothy J. Kobetz, Acting.

Dated at Rockville, Maryland, this 16th day of February, 2006.

For the Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. 06–1737 Filed 2–27–06; 8:45 am]

BILLING CODE 7590–01–P

[1] To the extent that the applications contain attachments and supporting documents that are not publicly available because they are asserted to contain safeguards or proprietary information, petitioners desiring access to this information should contact the applicant or applicant's counsel and discuss the need for a protective order.

POSTAL RATE COMMISSION

Briefings on International Mail and FY 2005 Cost and Revenue Analysis

AGENCY: Postal Rate Commission.

ACTION: Notice of briefings.

SUMMARY: The Commission will host two briefings on March 1, 2006. One will address a study of postal volume growth in developing countries. The other will address the effect of certain data collection design changes on a major Postal Service annual financial report. These briefings will provide an open forum for the presentation of information of interest to the postal community and the general public.

SUPPLEMENTARY INFORMATION: The first briefing will be presented by an economist in the Universal Postal Union's International Bureau, who will address the preliminary results of a study of factors that contribute to postal volume growth in developing countries. This briefing will also address the reasons why factors that affect postal volume growth in industrialized