Wednesday, March 3, 2004

9:30 a.m.

25th Anniversary Three Mile Island (TMI) Unit 2 Accident Presentation (Public Meeting) (Contact: Sam Walker, 301–415–1965)

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

2:45 p.m.

Discussion of Security Issues (Closed—Ex. 1)

Thursday, March 4, 2004

1:30 p.m.

Briefing on Status of Office of Nuclear Material Safety and Safeguards (NMSS) Programs, Performance, and Plans—Waste Safety (Public Meeting) (Contact: Claudia Seelig, 301–415–7243)

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

Week of March 8, 2004—Tentative

Tuesday, March 9, 2004

9:30 a.m.

Briefing on Status of Office of Nuclear Material Safety and Safeguards (NMSS) Programs, Performance, and Plans—Material Safety (Public Meeting) (Contact: Claudia Seelig, 301–415–7243)

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

1:30 p.m.

Discussion of Security Issues (Closed—Ex. 1)

The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: Timothy J. Frye, (301) 415–1651.

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

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: January 29, 2004.

Timothy J. Frye,

Technical Coordinator, Office of the Secretary.

[FR Doc. 04–2238 Filed 1–30–04; 10:12 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision 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, January 9, 2004, through January 22, 2004. The last biweekly notice was published on January 20, 2004 (69 FR 2735).

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.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By March 4, 2004, 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 2.714, 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 by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, 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 factors: (1) The nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of 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. Petitioner must provide 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 to relief. A petitioner who fails to file such a supplement which satisfies 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, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, 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 with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff, or may be delivered to the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. Because of continuing disruptions in delivery of mail to United States Government offices, it is requested that petitions for leave to intervene and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301–415–1101 or by e-mail to *hearingdocket@nrc.gov.* 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 because of continuing disruptions in delivery of mail to United States Government offices, 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 filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

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

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: December 3, 2003, as supplemented by letter dated January 14, 2004.

Description of amendment request: Appendix B, Additional Conditions, to Operating License No. DPR–23 for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, contains the following condition: ''Operation of H. B. Robinson Steam Electric Plant, Unit No. 2, is limited to 504 effective full power days. This additional condition shall remain in effect until approval of a license amendment that removes this limitation.'' The proposed change will delete the condition described above.

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:

An evaluation of the proposed change has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this amendment request follows:

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

The proposed change to Appendix B of the HBRSEP, Unit No. 2, Operating License deletes a restriction on effective full-power days (EFPD) that was incorporated to ensure the source term used for radiological dose analyses remains bounded by the analyses of record for operation at the approved, uprated power level. The restriction was imposed solely for the post-accident radiological analyses assumption. Since this restriction is only related to post-accident analytical assumptions, it is unrelated to the probability of an accident occurring. Therefore, the proposed Operating License change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed change can impact the consequences of previously evaluated accidents by impacting the core inventory of radionuclides for operating periods exceeding the existing 504 EFPD restriction. An evaluation of the potential impact of removing the EFPD restriction on the accident consequences has determined that any increase in consequences would be less than 10% of the difference between the existing dose analysis results and the acceptable dose limits. The proposed change therefore results in less than a minimal increase in accident consequences. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

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

The proposed change to Appendix B of the HBRSEP, Unit No. 2, Operating License deletes a restriction on effective full-power days (EFPD) that was incorporated to ensure the source term used for radiological dose analyses remain bounded by the dose analyses of record for operation at the approved, uprated power level. The restriction was imposed solely for post-accident radiological analyses assumptions. Since this restriction is only related to post-accident analytical assumptions, it is unrelated to the possibility of an accident occurring. Therefore, this 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 the Margin of Safety.

The applicable margin of safety is that related to the dose consequences of analyzed accidents. The proposed change results in potential increased consequences that are less than 10% of the difference between the existing dose analyses results and acceptable dose limits. This is less than a minimal increase in accident consequences, as defined by NEI [Nuclear Energy Institute] 96–07, Revision 1, which is endorsed by Regulatory Guide 1.187. Therefore, this change does not involve a significant reduction in a margin of safety.

Based on the above discussion, Progress Energy Carolinas, Inc. [Carolina Power & Light Company] has determined that the requested change does not involve a significant hazards consideration.

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

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

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: October 22, 2003.

Description of amendment request: The proposed amendment would delete requirements from the Technical Specifications (TSs) to maintain hydrogen recombiners and hydrogen and oxygen monitors. A notice of availability for this technical specification improvement using the consolidated line item improvement process (CLIIP) was published in the **Federal Register** on September 25, 2003 (68 FR 55416). Licensees were generally required to implement upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to combustible gas control were imposed by Order for many facilities and were added to or included in the TSs for nuclear power reactors currently licensed to operate. The revised 10 CFR 50.44, ''Standards for combustible gas control system in light-water-cooled power reactors,'' eliminated the requirements for hydrogen recombiners and relaxed safety classifications and licensee commitments to certain design and qualification criteria for hydrogen and oxygen monitors.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

NRC Section Chief: L. Raghavan.

Exelon Generation Company, LLC, Docket No. 50–352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of amendment request: December 22, 2003.

Description of amendment request: Exelon Generation Company, LLC, the licensee, is proposing a change to the Limerick Generating Station (LGS), Unit 1, Technical Specifications (TSs) contained in Appendix A to the Operating License. This proposed change will revise the TS section on safety limits to incorporate revised safety limit minimum critical power ratios (SLMCPRs) based on cycle-specific analysis performed by Global Nuclear Fuel for LGS, Unit 1, Cycle 11.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below.

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

Changing the SLMCPRs does not require any physical plant modifications, physically affect any plant components, or involve changes in plant operation. Therefore, the probability of an accident previously evaluated remains unchanged. The operability of plant systems designed to mitigate any consequences of accidents has not changed, therefore, the consequences of an accident previously evaluated are not expected to increase.

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

The proposed change does not involve any modifications of the plant configuration for allowable modes of operation. The SLMCPRs are not accident initiators, and their revision will 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?

The proposed SLMCPRs provide a margin of safety by ensuring that no more than 0.1% of the fuel rods are in a boiling transition if the operating limit minimum critical power ratios are exceeded during any mode of operation. Although the SLMCPRS are being reduced from 1.10 to 1.07 for two loop operation, and from 1.11 to 1.08 for single loop operation, the SLMCPRs continue to ensure that during normal operation and abnormal operational transients at least 99.9% of all fuel rods in the core do not experience transition boiling if the limit is not violated when all uncertainties are considered, thereby preserving the fuel cladding integrity. Therefore, the proposed TS change will not involve a significant reduction in a margin of safety.

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. Edward Cullen, Vice President & General Counsel, Exelon Generation Company, LLC, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: Darrell Roberts, Acting.

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

Date of amendment request: November 25, 2003.

Description of amendment request: Exelon Generation Company, LLC, the licensee, is proposing a change to the Limerick Generating Station (LGS), Units 1 and 2, Technical Specifications (TSs) contained in Appendix A to Operating Licenses NPF–39 and NPF–85, respectively. This proposed change will add a footnote to TS 3.4.3.2.e to indicate that reactor coolant system (RCS) pressure isolation valve (PIV) leakage is excluded from any other allowable RCS operational leakage specified in TS Section 3.4.3.2.

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. PIV leakage is not operational leakage. PIV leakage limits are used in conjunction with the system specifications for the PIVs to ensure that plant operation is appropriately limited. The

PIV leakage limit provides for monitoring the condition of the RCPB [Reactor Coolant Pressure Boundary] to detect PIV degradation. Although the proposed change will result in a change to the current method of calculating the RCS operational leakage, the proposed change does not affect the actual PIV leakage limit itself, and therefore, does not affect the ability to detect PIV degradation. The proposed change does not affect the basis for the safety analysis used to determine the probability or consequences of an accident since PIV leakage is not considered in any design basis accident.

Although the effect of the proposed change will allow for the potential increase in identified leakage, the total RCS operational leakage is still limited by the Technical Specifications (TS) Limiting Condition for Operation (LCO) which itself is not being changed. In addition, current TS Applicability, Action and Surveillance requirements for detection, monitoring, and appropriately limiting operational leakage are not being changed.

The proposed change does not alter the leakage detection system monitors, design features, operation, or accident analysis assumptions which could affect the ability of the reactor coolant pressure boundary (RCPB) to mitigate the consequences of a previously evaluated accident. The proposed change will not increase the likelihood of the malfunction of another system, structure or component which has been assumed as an accident initiator or credited in the mitigation of an accident.

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

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

Response: No. The operational leakage requirements for the RCPB leakage, unidentified leakage, and total leakage ensures corrective action can be taken to protect the RCPB from degradation. The PIV leakage provides an indication that the PIVs, between the RCS and the connecting systems, are degraded or degrading.

No change in the ability to perform the design function of the leak detection system, the protection afforded by the operational leakage requirements, or PIV leakage requirements is involved. No change in the operation of the leak detection system or PIVs is required. Instrumentation setpoints, monitoring frequencies and leakage limitations associated with RCS operational leakage and PIV leakage are not affected by the proposed change. No modifications to the PlVs or RCS leak detection system or associated components are required to implement the proposed change. Therefore, no new failure mechanism, malfunction, or accident initiator is considered credible.

Additionally, the proposed change does not affect other plant design, hardware, or system operation. 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 involve a relaxation of the criteria used to establish safety limits, a relaxation of the bases for the limiting safety system settings, or a relaxation of the bases for the limiting conditions for operation, other than excluding PIV leakage from the other RCS operational leakage.

Controlling values for the RCS operational leakage and PIV leakage are included in current TS testing measurements, monitors, detection methods and procedures. The proposed change will not modify these requirements or the accident analysis assumptions regarding the performance of the RCS operational leakage and PIV leakage monitoring which could potentially challenge safety margins established to ensure fuel cladding integrity, as well as reactor coolant and containment system integrity.

The safety analyses of the RCPB integrity and the ability to mitigate accidents do not require revision in order to implement the proposed change. Modification of the existing margins is not required.

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

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

Attorney for licensee: Mr. Edward Cullen, Vice President & General Counsel, Exelon Generation Company, LLC, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: Darrell Roberts, Acting.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of amendment request: October 14, 2003.

Description of amendment request: The proposed changes would eliminate License information that no longer applies to a license that has permanently ceased operation. The proposed changes would also simplify the Technical Specifications. Maine Yankee proposes to remove certain design and administrative requirements, relocate them to the Defueled Safety Analysis Report (DSAR) (i.e., Updated Final Safety Analysis Report for Maine Yankee), or the Quality Assurance Program and make other minor administrative changes. The DSAR is controlled by 10 CFR 50.59, and the Quality Assurance Program is controlled by 10 CFR 50.54(a). The Technical Specification relocation is being proposed pursuant to the criteria contained in 10 CFR 50.36, and is consistent with NRC Administrative Letter 95–06. Additionally, Maine Yankee proposes to eliminate technical specifications which will no longer be applicable following the transfer of the last fuel assembly from the spent fuel pool to spent fuel storage cask.

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 License changes delete License information that does not apply to a plant that has permanently ceased operation. These changes are in compliance with 10 CFR Part 50 regulations and are not associated with the probability or consequences of accidents previously evaluated.

The proposed Technical Specification changes reflect the complete transfer of all spent nuclear fuel from the Spent Fuel Pool (SFP) to the Independent Spent Fuel Storage Installation (ISFSI). Design basis accidents related to the Spent Fuel Pool are discussed in the MY Defueled Safety Analysis Report (DSAR). These postulated accidents are predicated on spent nuclear fuel being stored in the Spent Fuel Pool. With the removal of the spent fuel from the Spent Fuel Pool, there are no remaining safety related systems required to be monitored and there are no remaining credible design basis accidents related to the SFP.

The proposed relocation of the specified minimum distance to the Exclusion Area Boundary from the Technical Specification to the DSAR has no impact on the probability or consequences of the remaining applicable design basis accidents.

The proposed changes do not affect design functions of structures, systems or components (SSC's) associated with the safe storage of fuel or radioactive material. Nor do any of these changes increase the likelihood of the malfunction of an SSC. The proposed changes do not affect operating procedures or administrative controls that have the function of preventing or mitigating any design basis accidents.

The MY DSAR provides a discussion of radiological events postulated to occur as a result of decommissioning with the bounding consequence resulting from a materials handling event. The proposed changes do not have an adverse impact on decommissioning activities or any of their postulated consequences.

In addition, the proposed Technical Specification changes are consistent with the guidance provided in NRC Administrative Letter 95–06. Therefore, these proposed changes do not involve a significant increase in the probability or consequences of any 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 License changes delete License information that does not apply to a plant that has permanently ceased operation. These changes are in compliance with 10 CFR Part 50 regulations and are not associated with any accidents previously evaluated.

These proposed Technical Specification changes relocate requirements from the Technical Specifications to the Defueled Safety Analysis Report, eliminate Technical Specifications associated with the storage of spent fuel in the SFP, and relocate Technical Administrative Controls to the MY Quality Assurance Program. With the complete removal of spent fuel assemblies from the plant there are no safety related SSC's that remain at the plant. Thus, these proposed changes will not have any affect on the operation or design function of safety related SSC's. These changes do not create new component failure mechanisms, malfunctions or accident initiators. Therefore, these proposed changes would 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 License changes delete License information that does not apply to a plant that has permanently ceased operation. These changes are in compliance with 10 CFR Part 50 regulations and do not involve a reduction in a margin of safety.

The design basis and accident assumptions within the MY DSAR and the Defueled Technical Specifications relating to spent fuel are no longer applicable. The proposed Technical Specification changes do not affect remaining plant operations, systems, or components supporting decommissioning activities. In addition, the proposed changes do not result in a change in initial conditions, system response time, or in any other parameter affecting the course of a decommissioning activity accident analysis.

The relocation of the specified minimum distance to the Exclusion Area Boundary from the Technical Specifications to the Defueled Safety Analysis Report is consistent with the criterion set forth in 10 CFR 50.36 (c)(4). This criterion states that design features to be included in the Technical Specifications are those features of the facility such as materials of construction and geometric arrangement, which if altered or modified, would have a significant effect on safety and are not covered in other Technical Specification categories. The minimum distance to the Exclusion Area Boundary is established to maintain compliance within the limits specified in 10 CFR Part 100. The relocation of the specified minimum distance to the Exclusion Area Boundary to the DSAR continues to provide the safety analysis controls to assure compliance with 10 CFR Part 100 regulation.

Conclusion

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

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

Attorney for licensee: Joe Fay, Esquire, Maine Yankee Atomic Power Company, 321 Old Ferry Road, Wiscasset, Maine 04578.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment requests: December 19, 2003.

Description of amendment requests: The proposed license amendment request (LAR) would revise Technical Specification (TS) 5.5.9, ''Steam Generator (SG) Tube Surveillance Program.'' The LAR proposes new SG wedge region exclusion zones for outside diameter stress corrosion cracking (ODSCC) alternate repair criteria (ARC) at tube support plate (TSP) intersections and for primary water stress corrosion cracking (PWSCC) ARC at dented TSP intersections. The wedge region exclusion zones currently approved for the ODSCC ARC and for the PWSCC ARC are based on a loss-of-coolant accident (LOCA) plus safe shutdown earthquake (SSE) loads analysis performed in 1992. The new wedge region exclusion zones are based on new analyses of LOCA plus SSE loads completed in 2003 using plant-specific accident loads.

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

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

Application of a smaller steam generator (SG) wedge region exclusion zone will allow more degraded tubes to remain in service under alternate repair criteria (ARC). Previously approved ARC limits will be applied to tubes outside the exclusion zone, and therefore the probability and consequences of tube burst or leakage is not significantly increased following a steam line break (SLB).

Exclusion zones tubes are inspected by bobbin coil every outage and by rotating pancake coil (RPC) if the bobbin coil detects degradation. SG tubes containing RPC-confirmed crack-like degradation at wedge region exclusion zone intersections will be repaired. Because in-service tube intersections in wedge region exclusion zones do not have detectable cracking, they will not be susceptible to in-leakage if deformed following a loss-of-coolant-accident (LOCA) plus seismic event. Therefore, the consequences of a LOCA plus seismic event are not increased.

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

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

Implementation of revised SG ARC wedge region exclusion zones will allow more degraded tubes to remain in service under ARC. Implementation of ARC has been previously approved and does not introduce any significant change to the plant design basis. A single or multiple tube rupture event would not be expected in a SG in which ARC has been applied.

Therefore, the proposed changes do 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.

Revised wedge region exclusion zones are based on a DCPP-specific analysis for the combined effects of a LOCA and safe shutdown earthquake (SSE) loads. The number of wedge region tubes that are predicted to deform has been decreased when compared to the prior analysis, which used highly conservative assumptions. The revised analysis incorporates DCPP-specific LOCA and seismic loads that were not available when the prior analysis was performed. The revised analysis also yields conservative results, such that the number of tubes in the exclusion zone (262 per SG) bound the number of tubes predicted to deform (120 per SG). Tubes located in the revised wedge region exclusion zone will continue to be subject to enhanced eddy current inspection requirements and will be excluded from application of ARC. Thus, existing tube integrity requirements continue to be met for these tubes and there is no change to the dose contribution from tube leakage. Offsite and control room doses will continue to meet the appropriate guidelines and regulations established in Standard Review Plan 15.1.5 and 6.4, 10 CFR part 100, and 10 CFR part 50, Appendix A General Design Criterion (GDC) 19.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment requests: January 7, 2004.

Description of amendment requests: The license amendment request (LAR) would revise Sections 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' and 5.6.10, ''Steam Generator (SG) Tube Inspection Report'' of the Diablo Canyon Technical Specifications. The proposed changes would allow application of a 4-volt alternate repair criteria (ARC) in intersections of steam generator (SG) tube hot-legs with the four lowest SG tube support plates (TSPs). The 4-volt ARC will only apply to Model 51 SG tubes experiencing outside diameter stress corrosion cracking (ODSCC) at the intersections of the tube hot legs and the four lowest TSPs. In addition, the proposed change includes the application of leak-before-break (LBB) to the main steam line (MSL) piping inside containment in order to exclude the dynamic effects of a main steam line break (MSLB) in the short length of piping upstream of the MSL flow restrictor (large MSLB) from consideration for determining the loads on the SG TSPs following an MSLB.

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.

A 4-volt steam generator (SG) bobbin coil probe voltage-based alternate repair criteria (ARC) for axial outside diameter stress corrosion cracking (ODSCC) at tube support plate (TSP) locations is proposed for the hot-leg region of the SG tube at the 4 lowest TSP locations (TSPs 1 through 4). In order to implement the proposed 4-volt ARC, sufficient SG tubes will be expanded in the hot-leg region of TSPs 1 through 4 to limit the TSP deflections following a limiting main steam line break (MSLB) event.

SG tubes pass through holes drilled in the TSP. The inside diameter of the drilled holes closely approximates the outside diameter of the tubes. Generally, the TSP precludes those tube spans within the drilled holes from deforming beyond the diameters of the drilled holes, thus, precluding tube burst in the restrained regions. However, design basis MSLB events may vertically displace a TSP, removing its support from the tube spans passing through it. For TSPs at hot-leg locations in which sufficient SG tubes have been expanded at the TSP intersection, the deflections of the TSP following a limiting MSLB event are small, the TSPs remain essentially stationary during all conditions, and the SG tube spans within the drilled TSP holes are restrained. Thus, for intersections of SG tube hot-legs and TSPs 1 through 4, axial tube burst is eliminated as a credible event and the larger bobbin voltage for the proposed 4-volt ARC can be allowed while still meeting the tube structural requirements of Regulatory Guide (RG) 1.121 [, ''Bases for Plugging Degraded PWR Steam Generator Tubes''].

For the calculated displacement of the affected TSPs following a limiting design basis MSLB event, based on application of leak-before-break to the main steam system piping inside containment, tube hot-leg spans enclosed within TSPs 1 through 4 have a tube burst probability of much less than 10–5 collectively. This is orders of magnitude less than the 10–2 probability-of-burst criterion specified by Generic Letter (GL) 95–05 [, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking,] and represents negligible axial tube burst probabilities for affected tube hot-leg spans intersecting TSPs. Thus, repair limits to preclude burst are not needed and tube repair limits for intersections of SG tube hot-legs and TSPs 1 through 4 may be based primarily on limiting leakage to acceptable levels during accident conditions.

Cracks that include cellular corrosion may yield under axial loads, resulting in tensile tearing of the tube at that location. A tensile load requirement to prevent this establishes a structural limit for the tube expansion based ODSCC ARC. In order to establish a lower bound for the structural limit, tensile tests were used to measure the force required to separate a tube that exhibits cellular corrosion. Additionally, pulled SG tubes with cellular and/or inter-granular attack (IGA) tube wall degradation were evaluated and the tensile strength of the tube was conservatively calculated from the remaining noncorroded cross-section of the tube. The tensile strength calculation assumed that the degraded portions do not contribute to the axial load carrying ability of the tube. Data from these tests shows that circumferential cracks exhibiting bobbin coil probe indication voltages greater than 100 volts at the lower 95 percent confidence level require tube pressure differentials above the operating limit of 3-times normal operating differential pressure in order to produce circumferential ruptures (i.e., axial separation at the plane of the crack due to axial tensile tearing). The proposed 4-volt ARC has a safety factor of 25 to circumferential ruptures, which ensures the 4-volt ARC does not significantly increase the chances of a steam generator tube rupture (SGTR) at intersections of SG tube hot-legs and TSPs 1 through 4.

In addressing the potential combined Loss-of-coolant accident (LOCA) and earthquake effects on SG components as required by General Design Criterion (GDC) 2 of Appendix A to 10 CFR part 50, analysis has shown that SG tube deformation or collapse may occur in certain regions of the SG. SG tube collapse reduces RCS flow and could cause partial through-wall tube cracks to become full through-wall tube-cracks during tube deformation or collapse resulting in potential secondary-to-primary in-leakage to the reactor coolant system (RCS). Tubes for which deformation may occur are excluded from application of the voltage-based ARC per current TS 5.5.9.d.1.j (iv). TS 5.5.9.d.1.j (iv) will continue to apply and is not adversely affected by the 4-volt ARC. Therefore tubes for which deformation may occur will not be left in service under the 4-volt ARC.

GL 95–05 states that licensees must perform SG tube postaccident leak rate and SG tube burst probability analyses before returning to power from outages during which they perform SG inspections. Licensees must include the results in a report to the NRC within 90 days after restart. If an analysis reveals that postaccident leak-rate or burst-probability exceeds limits, the licensee must report it to the NRC and assess the safety significance of this finding.

For the proposed 4-volt ARC, the axial tensile tearing tube rupture probability is calculated for indications found at intersections of tube hot-legs and TSPs 1 through 4. The sum of MSLB axial tube burst probability for cold-leg TSP intersections, MSLB axial tube burst probability for hot-leg intersections at TSPs 5 through 7, axial tensile tearing tube rupture probability for TSPs 1 through 4, and the burst probability for indications left in service under other ARCs must be compared to the GL 95–05 reporting value of 10^{-2}. Due to the negligible burst probability for axial ODSCC indications at intersections of tube hot-legs and TSPs 1 through 4, calculation of the axial burst probability is not required for these indications.

The design basis MSLB outside of containment produces the limiting radiological consequence from any SG tube leakage due to SG tube indications that are postulated to exist at the initiation of an accident. Verification prior to each operating cycle, that the sum of MSLB leak rates from indications left in service under all ARC (including the proposed 4-volt ARC) are less than the leak rate limit assumed in the MSLB radiological consequences analysis, will ensure that site boundary doses for this accident remain within an acceptable fraction of the guidelines of Title 10 of the Code of Federal Regulations, Part 100, (10 CFR part 100) and that doses to the control room operators remain within the 10 CFR part 50, Appendix A GDC 19 limits.

The application of leak-before-break (LBB) to the MSL piping inside containment does not alter the way in which plant equipment is operated and cannot initiate an accident. The application of LBB to the main steam system does not affect the plant operating conditions and will not challenge the ability of the main steam system to perform its design function or to mitigate an accident.

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

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

Use of the proposed SG tube 4-volt ARC does not significantly change the operating conditions of the SG. Application of the 4-volt ARC does not significantly increase the probability of either single or multiple tube ruptures. SG tube integrity remains adequate for all plant operating conditions. The GL 95–05 SG tube integrity limits will be confirmed through in-service inspection and monitoring of primary-to-secondary leakage.

The Diablo Canyon Units 1 and 2 Technical Specifications (TS) impose a normal SG primary-to-secondary leak rate limit of 150 gallons per day (gpd) per SG to minimize the potential for excessive leakage during all plant conditions. The 150 gpd limit provides added margin to accommodate contingent leakage should a stress corrosion crack grow at a greater than expected rate or extend outside the TSP. The proposed 4-volt ARC does not adversely impact the TS 150 gpd limit. Normal operating leakage is not expected to significantly increase due to indications left in service under the proposed 4-volt ARC.

The application of LBB to the MSL piping inside containment does not involve any physical alteration to the plant or any change in which the plant is operated which could introduce a new failure mode. The use of LBB does not involve plant equipment being operated in a different manner.

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

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

RG 1.121 describes a method for meeting GDCs 14, 15, 31, and 32 of Appendix A to 10 CFR 50 by reducing the probability or consequences of SGTR through application of criteria for removing degraded tubes from service. These criteria set limits of degradation for SG tubing through inservice inspection. Analyses show that tube integrity will continue to meet the criteria of Regulatory Guide 1.121 after implementation of the proposed 4-volt ARC. Even under the worst case ODSCC occurrence at TSP elevations left in service under the 4-volt ARC, the 4-volt ARC will not cause or significantly increase the probability of a SGTR event.

Verification prior to each operating cycle, that the sum of MSLB leak rates from indications left in service under all ARC (including the proposed 4-volt ARC) are less than the leak rate limit assumed in the MSLB radiological consequences analysis, will ensure that site boundary doses for this accident remain within an acceptable fraction of the guidelines of 10 CFR 100 and that doses to the control room operators remain within the limits of GDC 19 of Appendix A to 10 CFR 50.

Inspections conducted for the proposed 4-volt ARC are the same as required by GL 95–05 with adjustment of the rotating pancake coil inspection requirements for hot-leg TSPs 1 through 4 intersections to reflect the higher 4-volt ARC limit. All hot-leg TSPs 1 through 4 intersections with bobbin coil voltages greater than 4 volts will be inspected with Plus Point coil and a Plus Point coil minimum sample inspection of intersections with bobbin indications less than or equal to 4 volts will be applied to hot-leg TSPs 1 through 4. The Plus Point coil data will be evaluated to confirm that the principal degradation mechanism continues to be ODSCC.

Plugging SG tubes reduces RCS flow margin. The 4-volt ARC will reduce the number of tubes that must be plugged. Thus, the 4-volt ARC will conserve RCS flow margin, preserving operational and safety benefits that would otherwise be reduced by unnecessary plugging.

The application of LBB to the MSL piping inside containment will not adversely affect operation of plant equipment and will not result in a change to design basis accident initial conditions or the setpoints at which protective actions are initiated. With application of LBB, the main steam system will continue to perform its function as assumed in the accident analyses.

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 Section Chief: Stephen Dembek.

Tennessee Valley Authority, Docket No. 50–259, Browns Ferry Nuclear Plant Unit 1, Limestone County, Alabama

Date of amendment request: November 3, 2003.

Description of amendment request: The proposed amendment would lower the allowable value for Function 7.b, Scram Discharge Volume Water Level—High Float Switches in Technical Specification (TS) Table 3.3.1.1–1, Reactor Protection System Instrumentation. As part of the proposed change, the licensee would also remove the Low Scram Pilot Air Header Pressure switches from service.

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

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

No. Modifications to the Scram Discharge Instrument Volume (SDIV) System are being implemented to ensure that the SDIV high water level instrumentation will respond adequately to provide redundant, diverse trip functions for a Scram Discharge Volume (SDV) inleakage event. The proposed change does not involve any change to the design or functional requirements of plant systems and the surveillance test methods will be unchanged. The proposed change will not give rise to any increase in operating power level, fuel operating limits, or effluents. The proposed change does not affect any accident precursors. In addition, the proposed change will not significantly increase any radiation levels. Since the scram function will be successfully performed, lowering the allowable value for the Scram Discharge Volume Water Level—High Float Switches and removal of the scram pilot air header pressure trip system does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

No. The design criteria for the Scram Discharge System is contained in the Safety Evaluation Report on the Boiling Water Reactor (BWR) Scram Discharge System, which was transmitted by NRC letter dated December 9, 1980, to All BWR Licensees. Modifications to the SDV System have been evaluated to demonstrate that the high water level instrumentation in the SDIV will respond adequately to provide the required trip function. No new system failure modes are created as a result of removing the low scram pilot air header trip, since the redundant and diverse SDIV high water level instruments will initiate a successful reactor scram. Therefore, lowering the allowable value for the Scram Discharge Volume Water Level—High Float Switches and removal of the scram pilot air header pressure trip system 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?

No. The water level in the SDIV is monitored by both resistance-temperature type detectors and float switches. Redundancy and diversity in the instrumentation that initiates the scram signal is maintained even with the lowering of the allowable value for the Scram Discharge Volume Water Level—High Float Switches and removal of the low scram pilot air header pressure trip function. Modifications to the SDIV System have been evaluated to demonstrate that the high water level instrumentation will respond adequately to provide the required trip function for an inleakage event. Therefore, the proposed amendment 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Section Chief: Allen G. Howe.

Tennessee Valley Authority, Docket No. 50–259, Browns Ferry Nuclear Plant Unit 1, Limestone County, Alabama

Date of amendment request: November 10, 2003.

Description of amendment request: The proposed amendment includes the necessary Technical Specification (TS) changes for the planned replacement of the power range monitoring portion of the existing Neutron Monitoring System with a digital upgrade. These changes would expand the current allowable operating domain to the Maximum Extended Load Line Limit region of the power/flow chart.

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.

Power Range Neutron Monitor (PRNM) Changes:

The proposed TS changes are associated with the Nuclear Measurement Analysis and Control (NUMAC) PRNM retrofit design. The proposed changes involve modification of the Limiting Conditions for Operation (LCOs) and Surveillance Requirements for equipment designed to mitigate events which result in power increase transients. For the Average Power Range Monitor (APRM) system, the mitigating action is to block control rod withdrawal or initiate a reactor scram which terminates the power increase when setpoints are exceeded. For the Rod Block Monitor (RBM) system, the mitigating action is to block continuous control rod withdrawal prior to exceeding the Minimum Critical Power Ratio safety limit during a postulated Rod Withdrawal Error event. The worst case failure of either the APRM or the RBM systems is failure to initiate its mitigating action (failure to scram or block rod withdrawal). Failure to initiate these mitigating actions will not increase the probability of an accident. Thus, the proposed changes do not increase the probability of an accident previously evaluated.

For the APRM and the RBM systems, the NUMAC PRNM design, together with revised operability requirements and revised surveillance requirements, results in equipment which continues to perform the same mitigation functions conditions with reliability equal to or greater than the equipment which it replaces. Because there is no change in mitigation functions and because reliability of the functions is maintained, the proposed changes do not involve an increase in the consequences of an accident previously evaluated.

APRM and RBM Technical Specification (ARTS) improvements and operation in an expanded core power/flow domain, the Maximum Extended Load Line Limit (MELLL) Changes:

The proposed ARTS/MELLL changes permit expansion of the current allowable power/flow operating region and will apply a newer methodology for assuring that fuel thermal and mechanical design limits are satisfied. Operation in the MELLL region with the ARTS changes has been evaluated and there is adequate design margin for operation in the MELLL region for all events and parameters considered. Because operation in the MELLL region maintains adequate design margin, the proposed changes do not increase the probability of an accident previously evaluated.

In support of operation in the MELLL region, the proposed change modifies flow-biased APRM scram and rod block setpoints and implements new RBM power-biased setpoints. No direct credit for the flow-biased APRM scram or APRM flow-biased rod block is taken in mitigation of any design basis event. Therefore, design margins are not degraded by the proposed changes.

The proposed changes to the RBM system will assure that a Rod Withdrawal Error is not a limiting event and that the RBM continues to enforce rod blocks under appropriate conditions.

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

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

Response: No.

The proposed PRNM and ARTS/MELLL changes involve modification and replacement of the existing power range neutron monitoring equipment, modification of the setpoints and operational requirements for the APRM and RBM systems, implementation of a new methodology for administering compliance with fuel thermal limits, and operation in an extended power/flow domain. These proposed changes do not modify the basic functional requirements of the affected equipment, create any new system interfaces or interactions, nor create any new system failure modes or sequence of events that could lead to an accident. The worst case failure of the affected equipment is failure to perform a mitigation action, and failure of this equipment to perform a mitigating action does not create the possibility of a new or different kind of accident. No new external threats or release pathways are created. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

PRNM Changes: These proposed TS changes are associated with the NUMAC PRNM retrofit design. The NUMAC PRNM change does not impact reactor operating parameters or the functional requirements of the PRNM system. The replacement equipment continues to provide information, enforce control rod blocks, and initiate reactor scrams under appropriate specified conditions. The proposed change does not revise any safety margin requirements. The replacement APRM/RBM equipment has improved channel trip accuracy compared to the current system, and meets or exceeds system requirements previously assumed in setpoint analysis. Thus, the ability of the new equipment to enforce compliance with margins of safety equals or exceeds the ability of the equipment which it replaces. Therefore, the proposed changes do not involve a reduction in a margin of safety.

ARTS/MELLL Changes: Operation in the MELLL region does not affect the ability of the plant safety-related trips or equipment to perform their functions, nor does it cause any significant increase in offsite radiation doses resulting from any analyzed event. Analyses have demonstrated that, for operation in the MELLL region, adequate margin to design limits is maintained. Implementation of the ARTS improvements provides flow- and power-dependent thermal limits which maintain existing margins of safety in normal operation, anticipated operational occurrences, and accident events. Implementation of power-biased RBM setpoints improves the margin of safety in a postulated Rod Withdraw Error (RWE) by assuring that the RWE is not a limiting event. Thus, the proposed changes do not involve a reduction in a margin of safety.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: December 18, 2003.

Brief description of amendments: The proposed amendments would revise Technical Specification (TS) 3.9.6, ''Residual Heat Removal (RHR) and Coolant Circulation Low Water Level'' to correct completion times for Actions B.2 and B.3. Action B.2 should have a completion time of immediately and Action B.3 should have a completion time of 4 hours.

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the Code of Federal Regulations (10 CFR), Section 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 change is considered to be a correction of an editorial error. The proposed revision to TS 3.9.6 is consistent with the current CPSES [Comanche Peak Steam Electric Station] licensing basis. Therefore the proposed 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.

The proposed change is considered to be an editorial correction and does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed change is considered to be an editorial correction and does not involve a reduction in a margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: December 31, 2003.

Brief description of amendments: The proposed amendments would revise Technical Specification (TS) 3.8.1, ''AC Sources—Operating'' to extend the allowable completion times for the required actions associated with restoration of an inoperable diesel generator (DG) and an inoperable offsite circuit (*i.e.*, startup transformer). The proposed amendments will also revise TS 3.8.9, ''Distribution Systems—Operating'' to extend the allowable completion times for the required actions associated with restoration of an inoperable alternating current (AC) electrical power distribution system (*i.e.*, 6.9 kV AC safety bus).

Basis for proposed no significant hazards consideration determination: As required by title 10 of the Code of Federal Regulations (10 CFR), Section 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 Technical Specification changes do not significantly increase the probability of occurrence of a previously evaluated accident because the 6.9 kV AC components (*i.e.*, Diesel Generators (DGs), startup transformers (STs), and safety-related (Class 1E) busses) are not initiators of previously evaluated accidents involving a loss of offsite power. The proposed changes to the Technical Specification Action Completion Times do not affect any of the assumptions used in the deterministic or the Probabilistic Safety Assessment (PSA) analysis[.]

The proposed Technical Specification changes will continue to ensure the 6.9 kV AC components perform their function when called upon. Extending the Technical Specification Completion Times to 10 days does not affect the design of the DGs, the operational characteristics of the DGs, the interfaces between the DGs and other plant systems, the function, or the reliability of the DGs. Thus, the DGs will be capable of performing either accident mitigation function and there is no impact to the radiological consequences of any accident analysis. To fully evaluate the effect of the changes to the 6.9 kV AC components, Probabilistic Safety Analysis (PSA) methods and deterministic analysis were utilized. The results of this analysis show no significant increase in the Core Damage Frequency.

The Configuration Risk Management Program (CRMP) in Technical Specification 5.5.18 is an administrative program that assesses risk based on plant status. Adding the requirement to implement the CRMP for Technical Specification 3.8.1 and 3.8.9 requires the consideration of other measures to mitigate consequences of an accident occurring while a 6.9 kV AC component is inoperable.

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

Response: No.

The proposed changes do not result in a change in the manner in which the electrical distribution subsystems provide plant protection. There are no design changes associated with the proposed changes. The changes to Completion Times do not change any existing accident scenarios, nor create any new or different accident scenarios.

The changes do not involve a physical alteration of the plant (*i.e.*, no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

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

Response: No.

The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not impacted by these changes. The proposed changes will not result in plant operation in a configuration outside the design basis. The calculated impact on risk is insignificant and is consistent with the acceptance criteria contained in Regulatory Guides 1.174 and 1.177. The proposed activities involve[] changes to certain Completion Times. The proposed changes remain bounded by the existing Surveillance Requirement Completion Times and therefore have no impact to the margins of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of application request: June 27, 2003, as revised by letter dated December 19, 2003, (previously published in the **Federal Register** on July 22, 2003 [68 FR 43396]).

Description of amendment request: The licensee's letter dated December 19, 2003, revises the original amendment application dated June 27, 2003. The original amendment request was described as that which would revise the Technical Specifications (TSs) to (1) extend the allowed outage time (AOT) or required action completion time (CT) for an inoperable diesel generator (DG) by adding the phrase ''*OR* 108 hours once per cycle for each DG'' to the completion time for Required Action B.4 in TS 3.8.1, ''AC Sources-Operating,'' and (2) delete the second CT given in certain required actions in TS 3.6.6, ''Containment Spray and Cooling Systems;'' TS 3.7.5, ''Auxiliary Feedwater (AFW) System;'' TS 3.8.1; and TS 3.8.9, ''Distribution System—Operating.'' The revised application dated December 19, 2003, requests changes to only Required Actions A.3 and B.4 for TS 3.8.1 to extend the AOT, or required action CT, for an inoperable DG.

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

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

The proposed changes for increasing the ''second'' Completion Times under TS 3.8.1 do not affect the design, operational characteristics, or intended functions of the equipment addressed by TS 3.8.1. With no direct effects on the subject equipment (or any other plant equipment or features), the proposed ''second'' Completion Time changes are not associated with any initiating condition for any accident previously evaluated, and therefore would not affect the probability of such accidents. Further, the consequences of evaluated accidents are independent of mitigating equipment allowed outage times as long as adequate availability of the equipment is ensured.

''Second'' Completion Times are primarily administrative in nature and are only intended to prevent successive, overlapping or contiguous entries and exits from Conditions within a Technical Specification LCO [Limiting Condition for Operation], which could otherwise result in an extended period of time for which the LCO is not met. The new, extended ''second'' Completion Times preserve this intent and were determined by the same method used to establish the original/existing second Completion Time limits, albeit with a longer, risk-informed Completion Time established for an inoperable diesel generator.

The proposed changes to the ''second'' Completion Times of TS 3.8.1 support the extended Completion Time/AOT specified for an inoperable diesel generator as proposed in AmerenUE's June 27, 2003 amendment application (Reference 1 [in AmerenUE's revised application dated December 19, 2003]). The acceptability and conformance to regulatory guidance for that change is addressed in [AmerenUE's June 27, 2003 amendment application], and the conclusions reached therein, including those reached with respect to significant hazards consideration, remain unchanged for that proposed change.

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

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

The proposed changes are primarily administrative in nature and do not involve a change in the design, configuration, or operational characteristics of the plant.

No physical alteration of the plant is involved, as no new or different type of equipment is to be installed. The changes do not alter any assumptions made in the safety analyses, and no alteration in the procedures for ensuring that the plant remains within analyzed limits is involved. As such, no new failure modes or mechanisms that could cause a new or different kind of accident from any previously evaluated are being introduced.

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

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

The proposed changes to the affected second Completion Times do not alter the manner in which safety limits or limiting safety system settings are determined. The safety analysis acceptance criteria are not impacted by [these] change[s], and the proposed changes will not permit plant operation in a configuration [that is] outside the design basis.

The proposed, extended second Completion Time limits were established in the same manner as the original limits, and meet the same intent, except that a longer risk-informed DG AOT has been used to establish the proposed second Completion Time limits[.] The basis and acceptability of that time limit is addressed in the June 27, 2003 amendment application (as supported by this supplemental/revision[, dated December 19, 2003]), and the conclusions reached therein still apply, including those reached with respect to [no] significant hazards consideration. [The June 27, 2003, amendment application stated: ''Further, with regard to plant risk, the risk assessment performed for the DG AOT extension determined that the quantifiable increase in plant risk is acceptably small.'']

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of application request: December 15, 2003.

Description of amendment request: The amendment would revise Technical Specifications (TSs) 3.3.9, ''Boron Dilution Mitigation System (BDMS),'' and 3.9.2, ''Unborated Water Source Isolation Valves.'' The proposed changes would replace the phrase ''unborated water'' by the word ''dilution'' in several places and delete references to isolation valves BGV0178 and BGV0601. A Note would also be added to TS 3.9.2 about dilution source path valves may be unisolated.

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

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

The proposed changes do not involve a significant increase in the probability or consequences of an inadvertent boron dilution accident by isolating the BTRS [boron thermal regeneration system] anion resin vessels in MODE 6 or by isolating the purge line for detector SJRE001 during flushing activities in MODE 6. By recognizing these potential dilution sources and by making TS 3.3.9 and TS 3.9.2 more generic for consideration of all potential dilution sources, plant administrative controls are revised such that the plant is put in a safer condition than before. Specific isolation [valve numbers] are removed from TS 3.3.9 and TS 3.9.2. They are relocated from the [Technical] Specifications to the appropriate TS Bases. This is an administrative only change and is consistent with the [Improved] Standard Technical Specifications, NUREG–1431[, that the Callaway Technical Specifications are based upon]. Allowing a dilution source path to be unisolated under administrative controls, described in TS Bases 3.9.1 during refueling decontamination activities, is acceptable as allowed by Amendment [No.] 97 to the Callaway Operating License and does not involve a significant increase in the probability or consequences of an inadvertent boron dilution accident.

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

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

The proposed changes do not create the possibility of a new or different kind of accident. Although other potential dilution sources are identified for administrative control, the evaluation of a MODE 6 dilution event remains unchanged. Isolating the BTRS anion vessels or isolating the purge line for detector SJRE001 during flushing activities in MODE 6 and making TS 3.3.9 and TS 3.9.2 more generic does not impact the operability of any safety related equipment required for plant operation. No new equipment will be added and no new limiting single failures are created. The plant will continue to be operated within the envelope of the existing safety analysis. In addition specific isolation [valve numbers] are removed from TS 3.3.9 and TS 3.9.2. They are relocated from the [Technical] Specifications to the appropriate TS Bases. This is an administrative only change and is consistent with the [Improved] Standard Technical Specifications, NUREG–1431 [, that the Callaway Technical Specifications are based upon]. Allowing a dilution source path to be unisolated under administrative controls, described in TS Bases 3.9.1 during refueling decontamination activities, is acceptable as allowed by Amendment [No.] 97 to the Callaway Operating License and does not create the possibility of a new or different kind of inadvertent boron dilution accident.

Therefore, the proposed changes do 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 changes do not reduce the margin of safety. Although other potential dilution sources are identified for administrative control and TS 3.3.9 and TS 3.9.2 are made more generic for consideration of all potential dilution sources, the evaluated margin of safety for a dilution event in MODE 6 remains the same. Recognition of other potential dilution sources, isolation of the BTRS anion resin beds and the purge line for detector SJRE001 during flushing activities in MODE 6, places the plant in a safer condition than before. In addition specific isolation [valve numbers] are removed from TS 3.3.9 and TS 3.9.2. They are relocated from the [Technical] Specifications to the appropriate TS Bases. This is an administrative only change and is consistent with the [Improved] Standard Technical Specifications, NUREG–1431 [, that the Callaway Technical Specifications are based upon]. Finally, allowing a dilution source path to be unisolated under administrative controls, described in TS Bases 3.9.1 during refueling decontamination activities, is acceptable as allowed by Amendment [No.] 97 to the Callaway Operating License and does not involve a significant reduction in a margin of safety due to an inadvertent boron dilution event.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of application request: December 17, 2003.

Description of amendment request: The amendment would revise Technical Specifications (TSs) 3.3.1, ''Reactor Trip System (RTS) Instrumentation,'' 3.3.2, ''Engineered Safety Feature Actuation System (ESFAS) Instrumentation,'' and 3.3.9, ''Boron Dilution Mitigation System (BDMS).'' The purpose of the amendment is to adopt the completion time, test bypass time, and surveillance frequency time changes approved by the NRC in Topical Reports WCAP–14333–P–A, ''Probabilistic Risk Analysis of the RPS [reactor protection system] and ESFAS Test Times and Completion Times,'' and WCAP–15376–P–A, ''Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times.'' The proposed changes would revise the required actions for certain action conditions; increase the completion times for several required actions (including some notes); delete notes in certain required actions; increase frequency time intervals (including certain notes) in several surveillance requirements (SRs); add an action condition and required actions; revise notes in certain SRs; and revise Table 3.3.2–1. There are also several administrative corrections to the format of the TSs (*e.g.*, moving the ''AND'' in the required actions for Condition O in TS 3.3.2).

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

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The same reactor trip system (RTS) and engineered safety feature actuation system (ESFAS) instrumentation will continue to be used. The protection systems will continue to function in a manner consistent with the plant design basis. These changes to the Technical Specifications [in the amendment] do not result in a condition where the design, material, and construction standards that were applicable prior to the change are altered.

The proposed changes will not modify any system interface. The proposed changes will not affect the probability of any event initiators [because the proposed changes are not 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 changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR [Callaway Final Safety Analysis Report].

The determination that the results of the proposed changes are acceptable [to be considered for plant-specific Technical Specifications] was established in the NRC Safety Evaluations prepared for WCAP–14333–P–A (issued by letter dated July 15, 1998) and for WCAP–15376–P–A (issued by letter dated December 20, 2002). Implementation of the proposed changes will result in an insignificant risk impact. Applicability of these conclusions has been verified through plant-specific reviews and implementation of the generic analysis results in accordance with the respective NRC Safety Evaluation conditions [for the two WCAPs].

The proposed changes to the Completion Times, test bypass times, and Surveillance Frequencies reduce the potential for inadvertent reactor trips and spurious ESF [engineered safety feature] actuations, and therefore do not increase the probability of any accident previously evaluated. The proposed changes do not change the response of the plant to any accidents and have an insignificant impact on the reliability of the RTS and ESFAS signals. The RTS and ESFAS will remain highly reliable and the proposed changes will not result in a significant increase in the risk of plant operation. This is demonstrated by showing that the impact on plant safety as measured by the increase in core damage frequency (CDF) is less than 1.0E–06 per year and the increase in large early release frequency (LERF) is less than 1.0E–07 per year. In addition, for the Completion Time changes, the incremental conditional core damage probabilities (ICCDP) and incremental conditional large early release probabilities (ICLERP) are less than 5.0E–07 and 5.0E–08, respectively. These changes meet the acceptance criteria in Regulatory Guides 1.174 and 1.177. Therefore, since the RTS and ESFAS will continue to perform their [safety] functions with high reliability as originally assumed, and the increase in risk as measured by ΔCDF, ΔLERF, ICCDP, ICLERP risk metrics is within the acceptance criteria of existing [NRC] regulatory guidance, there will not be a significant increase in the consequences of any accidents.

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended [safety] function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed changes are consistent with safety analysis assumptions and resultant consequences.

Therefore, [the] change[s do] not increase the probability or consequences of an accident previously evaluated.

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

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The proposed changes will not affect the normal method of plant operation. No performance requirements will be affected or eliminated. The proposed changes will not result in physical alteration to any plant system nor will there be any change in the method by which any safety-related plant system performs its safety function. There will be no setpoint changes or changes to accident analysis assumptions.

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

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

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

The proposed changes do not affect the acceptance criteria for any analyzed event nor is there a change to any Safety Analysis Limit (SAL). There will be no effect on the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, DNBR [departure from nucleate boiling ratio] limits, F_Q [heat flux hot channel factor], FΔH [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 will continue to be met.

Redundant RTS and ESFAS trains are maintained, and diversity with regard to the signals that provide reactor trip and engineered safety features actuation is also maintained. All signals credited as primary or secondary, and all operator actions credited in the accident analyses will remain the same. The proposed changes will not result in plant operation in a configuration outside the design basis. The calculated impact on risk is insignificant and meets the acceptance criteria contained in Regulatory Guides 1.174 and 1.177. Although there was no attempt to quantify any positive human factors benefit due to increased Completion Times and bypass test times, it is expected that there would be a net benefit due to a reduced potential for spurious reactor trips and actuations associated with testing.

Implementation of the proposed changes is expected to result in an overall improvement in safety, as follows:

(a) Reduced testing will result in fewer inadvertent reactor trips, less frequent actuation of ESFAS components, less frequent distraction of operations personnel without significantly affecting RTS and ESFAS reliability.

(b) Improvements in the effectiveness of the operating staff in monitoring and controlling plant operation will be realized. This is due to less frequent distraction of the operators and shift supervisor to attend to instrumentation Required Actions with short Completion Times.

(c) Longer repair times associated with increased Completion Times will lead to higher quality repairs and improved reliability.

(d) The Completion Time extensions for the reactor trip breakers will provide the utilities additional time to complete test and maintenance activities while at power, potentially reducing the number of forced outages related to compliance with reactor trip breaker Completion Times, and provide consistency with the Completion Times for the logic trains.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: December 15, 2003.

Description of amendment request: The licensee is proposing to revise Technical Specification (TS) Section 3.3.1, ''Reactor Trip System (RTS)

Instrumentation,'' and TS 3.3.2, ''Engineered Safety Feature Actuation System (ESFAS) Instrumentation,'' to adopt completion time, test bypass time (in Notes for several Required Actions), and surveillance frequency changes approved by the NRC in WCAP–14333–P–A, Revision 1, ''Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times,'' dated October 1998, and WCAP–15376–P–A, Revision 1, ''Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times,'' dated March 2003.

As part of this amendment, for TS 3.3.1, the Required Actions for Condition D, one power range neutron flux-high channel inoperable, are revised, and a Note for the Required Actions for Condition R is deleted.

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.

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The same reactor trip system (RTS) and engineered safety feature actuation system (ESFAS) instrumentation will continue to be used. The protection systems will continue to function in a manner consistent with the plant design basis. These changes to the Technical Specifications do not result in a condition where the design, material, and construction standards that were applicable prior to the change are altered.

The proposed changes will not modify any system interface. The proposed changes 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 changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the USAR [Updated Safety Analysis Report].

The determination that the results of the proposed changes are acceptable was established in the NRC Safety Evaluations prepared for WCAP–14333–P–A (issued by letter dated July 15, 1998) and for WCAP–15376–P–A (issued by letter dated December 20, 2002). Implementation of the proposed changes will result in an insignificant risk impact. Applicability of these conclusions has been verified through plant-specific reviews and implementation of the generic analysis results in accordance with the respective NRC Safety Evaluation conditions.

The proposed changes to the Completion Times, test bypass times, and Surveillance Frequencies reduce the potential for inadvertent reactor trips and spurious ESF [engineered safety feature] actuations, and therefore do not increase the probability of any accident previously evaluated. The proposed changes do not change the response of the plant to any accidents and have an insignificant impact on the reliability of the RTS and ESFAS signals. The RTS and ESFAS will remain highly reliable and the proposed changes will not result in a significant increase in the risk of plant operation. This is demonstrated by showing that the impact on plant safety as measured by the increase in core damage frequency (CDF) is less than 1.0E–06 per year and the increase in [the] large early release frequency (LERF) is less than 1.0E–07 per year. In addition, for the Completion Time changes, the incremental conditional core damage probabilities (ICCDP) and incremental conditional large early release probabilities (ICLERP) are less than 5.0E–07 and 5.0E–08, respectively. These changes meet the acceptance criteria in Regulatory Guides 1.174 and 1.177. Therefore, since the RTS and ESFAS will continue to perform their functions with high reliability as originally assumed, and the increase in risk as measured by ΔCDF, ΔLERF, ICCDP, ICLERP risk metrics is within the acceptance criteria of existing regulatory guidance, there will not be a significant increase in the consequences of any accidents.

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do 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 changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed changes are consistent with safety analysis assumptions and resultant consequences.

Therefore, [the proposed changes do] not increase the probability or consequences of an accident previously evaluated.

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

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The proposed changes will not affect the normal method of plant operation. No performance requirements will be affected or eliminated. The proposed changes will not result in [a] physical alteration to any plant system nor will there be any change in the method by which any safety-related plant system performs its safety function. There will be no setpoint changes or changes to accident analysis assumptions.

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

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

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

The proposed changes do not affect the acceptance criteria for any analyzed event nor is there a change to any Safety Analysis Limit (SAL). There will be no effect on the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, DNBR limits, F_Q, FΔH, LOCA [loss-of-coolant accident] PCT [peak cladding temperature], peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

Redundant RTS and ESFAS trains are maintained, and diversity with regard to the signals that provide reactor trip and engineered safety features actuation is also maintained. All signals credited as primary or secondary, and all operator actions credited in the accident analyses will remain the same. The proposed changes will not result in plant operation in a configuration outside the design basis. The calculated impact on risk is insignificant and meets the acceptance criteria contained in Regulatory Guides 1.174 and 1.177. Although there was no attempt to quantify any positive human factors benefit due to increased Completion Times and bypass test times, it is expected that there would be a net benefit due to a reduced potential for spurious reactor trips and actuations associated with testing.

Implementation of the proposed changes is expected to result in an overall improvement in safety, as follows:

(a) Reduced testing will result in fewer inadvertent reactor trips, less frequent actuation of ESFAS components, [and] less frequent distraction of operations personnel without significantly affecting RTS and ESFAS reliability.

(b) Improvements in the effectiveness of the operating staff in monitoring and controlling plant operation will be realized. This is due to less frequent distraction of the operators and shift supervisor to attend to instrumentation Required Actions with short Completion Times.

(c) Longer repair times associated with increased Completion Times will lead to higher quality repairs and improved reliability.

(d) The Completion Time extensions for the reactor trip breakers will provide the utilities additional time to complete test and maintenance activities while at power, potentially reducing the number of forced outages related to compliance with reactor trip breaker Completion Times, and provide consistency with the Completion Times for the logic trains.

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

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

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

NRC Section Chief: Stephen Dembek.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: January 29, 2003, as supplemented by letter dated September 15, 2003.

Brief description of amendment: The amendment proposes a one-time Technical Specification change to extend the test interval for the next Appendix J Type A test and the next drywell bypass leakage rate test from 10 to 15 years.

Date of issuance: January 8, 2004.

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

Amendment No.: 160.

Facility Operating License No. NPF–62: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** June 10, 2003 (68 FR 34661).

The supplemental letter of September 15, 2003, 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 January 8, 2004.

No significant hazards consideration comments received: No.

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: May 29, 2003.

Brief description of amendments: The amendments revised the Updated Final Safety Analysis Report to implement the Boiling Water Reactor Vessel and Internals Project reactor pressure vessel integrated surveillance program as the basis for demonstrating compliance with the requirements of Appendix H to 10 CFR part 50.

Date of issuance: January 14, 2004.

Effective date: As of the date of issuance.

Amendment Nos.: 229 and 257.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments revised the Updated Final Safety Analysis Report.

*Date of initial notice in **Federal Register:*** August 19, 2003 (68 FR 49814).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 14, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 10, 2003, as supplemented December 30, 2003.

Brief description of amendment: The amendment revises Technical Specification (TS) 3.7.3, ''Control Room Emergency Filtration (CREF) System,'' Surveillance Requirement (SR) 3.7.3.6, to permit a one-time deferral of SR 3.7.3.6 until startup from the next refueling outage (RF–10) to preclude a mid-cycle shutdown solely for the performance of this SR. SR 3.7.3.6 requires verifying that unfiltered in-leakage from CREF system duct work outside the control room envelope that is at negative pressure during accident conditions is within limits. This SR is required to be performed every 36 months, and can be performed only when the CREF system is not required to be OPERABLE (*i.e.*, in MODES 4 or 5, with no operations with a potential for draining the reactor vessel and with no fuel movement of recently irradiated fuel in progress).

Date of issuance: January 16, 2004.

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

Amendment No.: 158.

Facility Operating License No. NPF–43: Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** November 25, 2003 (68 FR 66134).

The December 30, 2003, supplemental letter provided additional clarifying information that was within the scope of the original application and did not change the Nuclear Regulatory Commission staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 16, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 24, 2003, as supplemented by letters dated June 25 and October 15, 2003.

Brief description of amendments: The amendments revise the Technical Specifications (TS) to relocate certain reactor coolant system cycle-specific parameter limits from the TSs to the Core Operating Limits Report, and revises the minimum allowable reactor coolant system flow rate.

Date of issuance: January 14, 2004.

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

Amendment Nos.: 219 and 201.

Renewed Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** September 18, 2003 (68 FR 54749), November 18, 2003 (68 FR 65090).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 14, 2004.

No significant hazards consideration comments received: No.

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

Date of application of amendments: November 14, 2002, as supplemented by letter dated April 14, 2003.

Brief description of amendments: The amendments revised the Technical Specification 3.3.1 ''Reactor Protective System (RPS) Instrumentation,'' Surveillance Requirement 3.3.1.3 to add a correlation slope to the formula for axial power imbalance error.

Date of Issuance: January 15, 2004.

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

Amendment Nos.: 337, 337 and 338.

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

Date of initial notice in ***Federal Register:*** December 10, 2002 (67 FR 75870).

The supplement dated April 14, 2003, provided clarifying information that did not change the scope of the November 14, 2002, application nor the initial proposed no significant hazards consideration determination.

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

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: July 1, 2003, as supplemented December 10, 2003.

Brief description of amendments: The amendments revise Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF–11 and NPF–18. Specifically, the changes delete one and add two references to the list of analytical methods in TS 5.6.5, ''Core Operating Limits Report (COLR),'' that can be used to determine core operating limits. The deleted reference is to an analytical method that is no longer applicable to LaSalle County Station (LSCS). The new references will allow LSCS to use General Electric Company (GE) methods for the determination of fuel assembly critical power of Framatome Advanced Nuclear Fuel, Inc. (Framatome) Atrium-9B and Atrium-10 fuel. The changes are the result of a LSCS decision to insert GE14 fuel during the upcoming refueling outage at LSCS Unit 1 in January 2004. GE's safety analysis methodologies have been previously used at LSCS and GE14 fuel is currently in use at other Exelon Generation Company, LLC (Exelon), stations.

The first added reference, ''GEXL96 Correlation for Atrium-9B Fuel,'' lists a method that was previously approved by the NRC for use by licensees. The second added reference, ''GEXL97 Correlation for Atrium-10 Fuel,'' lists a GE method for determining the critical power for Atrium-10 fuel. This correlation had not been previously reviewed and approved by the NRC for use by licensees. Additionally, editorial changes are made to existing references.

Date of issuance: January 9, 2004.

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

Amendment Nos.: 164 and 150.

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

Date of initial notice in ***Federal Register:*** November 12, 2003 (68 FR 64135). The supplement dated December 10, 2003, provided clarifying information that did not change the scope of the July 1, 2003, application nor the initial no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

GPU Nuclear Corporation and Saxton Nuclear Experimental Corporation (SNEC), Docket No. 50–146, Saxton Nuclear Experimental Facility (SNEF)

Date of application for amendment: April 22, 2002, as supplemented on December 5, 2002, and September 30 and December 22, 2003.

Brief description of amendment: The amendment allows removal of the upper half of the SNEF containment vessel and makes a change to the organization to add the position of Vice-President GPU Nuclear Oversight to reflect the merger of GPU Inc. and FirstEnergy Corp.

Date of Issuance: January 9, 2004.

Effective Date: January 9, 2004.

Amendment No.: 19.

Amended Facility License No. DPR–4: Amendment changed the Technical Specifications.

Date of initial notice in the ***Federal Register:*** January 7, 2003, with a correction notice published on January 22, 2003. The letters of September 30 and December 22, 2003, supplied clarifying information that did not expand the scope of the January 5, 2003, or January 22, 2003, **Federal Register** Notices. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 9, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 17, 2002, as supplemented December 10, 2003.

Brief description of amendment: The amendment revises Technical Specification Table 3.3.1–2 by modifying a constant in the variable thermal margin/low pressure (TM/LP) trip equation. The change reduces calculated values for the variable TM/LP trip setpoint, and results from improvements in plant equipment used to establish the TM/LP trip setpoint. Ultrasonic feedwater flow measurement devices, which were recently installed at Palisades, result in less uncertainty applied in the methodology used for determining core power level. The devices used to calculate the TM/LP trip setpoint were previously replaced with digital thermal margin monitors having less uncertainty.

Date of issuance: January 8, 2004.

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

Amendment No.: 214.

Facility Operating License No. DPR–20. Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 2, 2003 (68 FR 52235).

The December 10, 2003, letter provided additional information in support of the initial application, did not expand the scope of the application as originally noticed, and did not effect the NRC's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 8, 2004.

No significant hazards consideration comments received: No.

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

Date of amendment request: July 18, 2003, as revised by letter dated August 28, 2003, and supplemental letters dated October 31 and December 15, 2003.

Brief description of amendment: The amendment revises the renewed operating license and technical specifications to increase the licensed rated power by 1.6 percent from 1500 megawatts thermal (MWt) to 1524 MWt.

Date of issuance: January 16, 2004.

Effective date: January 16, 2004, and shall be implemented within 30 days of the date of issuance. Modifications associated with the measurement uncertainty recapture power uprate will be completed prior to implementation. This includes: (1) Implementation of control room alarm functions, and (2) Figure 2–1 of the Pressure-Temperature Limits Report will be revised prior to the reactor vessel reaching 39.9 effective full power years of operation.

Amendment No.: 224.

Renewed Facility Operating License No. DPR–40: The amendment revised the Operating License and Technical Specifications.

*Date of initial notice in **Federal Register:*** September 18, 2003 (68 FR 54751).

The October 31 and December 15, 2003, supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50–366, Edwin I. Hatch Nuclear Plant, Unit 2, Appling County, Georgia

Date of application for amendment: December 4, 2002, as supplemented by letters dated June 24, and October 23, 2003.

Brief description of amendment: The amendment revised the Technical Specification regarding the turbine building high temperature primary containment isolation value specified in Table 3.3.6.1–1, Item 1f.

Date of issuance: January 12, 2004.

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

Amendment No.: 181.

Renewed Facility Operating License No. NPF–5: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 21, 2003 (68 FR 2807).

The supplements dated June 24 and October 23, 2003, provided clarifying information that did not change the scope of the December 4, 2002, application nor the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: February 26, 2003, as supplemented by letter dated July 25, 2003.

Brief description of amendments: The amendments revised the Technical Specifications Section 5.5.17, ''Containment Leakage Rate Testing Program,'' to reflect a one time deferral of the Type-A Containment Integrated Leak Rate Test (ILRT). The 10-year interval between ILRTs is to be extended to 15 years from the previous ILRTs that were completed in March 2002 for Unit 1 and March 1995 for Unit 2.

Date of issuance: January 12, 2004.

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

Amendment Nos.: 130 and 108.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 13, 2003 (68 FR 25658).

The supplement dated July 25, 2003, provided clarifying information that did not change the scope of the February 26, 2003, application nor the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 26th day of January 2004.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. 04–2017 Filed 2–2–04; 8:45 am]

BILLING CODE 7590–01–P

RAILROAD RETIREMENT BOARD

Agency Forms Submitted for OMB Review

SUMMARY: In accordance with the Paperwork Reduction Act of 1995 (44 U.S.C. Chapter 35), the Railroad Retirement Board (RRB) has submitted the following proposal(s) for the collection of information to the Office of Management and Budget for review and approval.

Summary of Proposal(s)

(1) *Collection title:* Railroad Unemployment Insurance Act Applications.

(2) *Form(s) submitted:* SI–1a, SI–1b, SI–3, SI–7, SI–8, ID–7H, ID–11A, ID–11–B.

(3) *OMB Number:* 3220–0039.

(4) *Expiration date of current OMB clearance:* 5/31/2004.

(5) *Type of request:* Revision of a currently approved collection.

(6) *Respondents:* Individuals or households, Business or other for-profit.

(7) *Estimated annual number of respondents:* 44,600.

(8) *Total annual responses:* 260,900.

(9) *Total annual reporting hours:* 26,321.

(10) *Collection description:* Under section 2 of the Railroad Unemployment Insurance Act, sickness benefits are payable to qualified railroad employees who are unable to work because of