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

Week of February 10, 2003—Tentative

Monday, February 10, 2003

10 a.m. Briefing on Status of Office of Nuclear Reactor Regulation (NRR) Programs, Performance, and Plans (Public Meeting) (Contact: Michael Case, 301–415–1275)

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

Tuesday, February 11, 2003

10 a.m. Briefing on Status of Office of the Chief Financial Officer (OCFO) Programs, Performance, and plans (Public Meeting) (Contact: Lars Solander, 301–415–6080)

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

*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: David Louis Gamberoni (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 2, 2003.

David Louis Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 03–320 Filed 1–3–03; 12:17 pm]

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, December 13, through December 26, 2002. The last biweekly notice was published on December 24, 2002 (67 FR 78515).

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

By February 6, 2003, 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,[1] which is available at the Commission's PDR, located at One White Flint North, 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

[1] The most recent version of Title 10 of the Code of Federal Regulations, published January 1, 2002, inadvertently omitted the last sentence of 10 CFR 2.714 (d) and paragraphs (d)(1) and (d)(2) regarding petitions to intervene and contentions. For the complete, corrected text of 10 CFR 2.714 (d), please see 67 FR 20884; April 29, 2002.

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, 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, 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, 304–415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of amendment request: November 13, 2002, as supplemented November 20, 2002

Description of amendment request: The proposed amendments delete requirements from the technical specifications (TS) and other elements of the licensing bases to maintain a Post-Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''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 PASS were imposed by Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The changes are based on NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–413, ''Elimination of Requirements for a Post-Accident Sampling System (PASS).'' The NRC staff issued a notice of opportunity for comment in the **Federal Register** on December 27, 2001 (66 FR 66949), on possible amendments concerning TSTF–413, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 20, 2002 (67 FR 13027). The licensee affirmed the applicability of the following NSHC determination in its application dated November 13, 2002.

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

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

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

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

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

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

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

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

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

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

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

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

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

Attorney for licensee: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: December 2, 2002.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) for Administrative Controls in Section 5.0 concerning Responsibility, Unit Staff, Unit Staff Qualifications, and Controls of the High Radiation Area.

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:

As required by 10 CFR 50.91(a)(1), this analysis is provided to demonstrate that the proposed license amendment does not involve a significant hazard.

Conformance of the proposed amendment to the standards for a determination of no significant hazards, as defined in 10 CFR 50.92, is shown in the following:

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

No. Implementation of this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. Approval of this amendment will have no effect on accident probabilities or consequences since the changes are purely administrative in nature.

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

No. Implementation of this amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of NRC approval of this amendment request. No physical changes are being made to the plant. Therefore, the introduction of any new accident scenarios does not exist. The amendment does not impact any plant systems that are accident initiators nor does it adversely impact any accident mitigating system. This amendment is purely administrative in nature.

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

No. Implementation of this amendment will not involve a significant reduction in a margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these fission product barriers will not be impacted by implementation of this amendment. System[s] and components are not affected and therefore are capable of performing as designed. This amendment is purely administrative nature, it will have no effect on any safety margins.

Conclusion.

Based on the preceding analysis, it is concluded that the proposed license amendment does not involve a Significant

Hazards Consideration Finding as defined in 10 CFR 50.92.

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: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: December 12, 2002.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) for TS Table 3.3.2–1 Footnote (c) to correct an editorial error, TS 3.4.3 is revised to update the Reactor Coolant System Pressure-Temperature limits for use up to 34 Effective Full Power Years (EFPY) and TS 3.4.12 is revised to update the Low Temperature Over-Pressure limits for use up to 34 EFPY. Associated changes are also proposed for the TS Bases.

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:

Duke has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, ''issuance of amendment,'' as discussed below:

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

Response: No.

The proposed changes to the reactor coolant system (RCS) pressure and temperature (P–T) limits and low temperature overpressure protection (LTOP) limits are developed utilizing the methodology of American Society of Mechanical Engineers (ASME) Section XI, Appendix G, in conjunction with the methodology of ASME Code Case N–641. Usage of these methodologies provides compliance with the underlying intent of 10 CFR [Part] 50 Appendix G and provides operational limits established to prevent non-ductile failure of the reactor vessel. The Loss of Coolant Accident analysis and other accident analyses in the Updated Final Safety Analysis Report (UFSAR) do not assume failure of the reactor vessel. The P–T and LTOP limits are not initiators or contributors to accident analyses addressed in the UFSAR. The proposed changes do not alter any assumption previously made in the radiological consequence evaluations nor affect the mitigation of the radiological consequences of an accident previously evaluated. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The changes to RCS P–T limits and LTOP limits are proposed to prevent non-ductile failure of the reactor vessel. The proposed changes do not modify the RCS pressure boundary, nor make any physical changes to the facility. The proposed changes do not introduce any new mode of system operation or failure mechanism. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed changes are developed utilizing the methodology of ASME Section XI, Appendix G, in conjunction with the methodology of ASME Code Case N–461. Usage of these methodologies provides compliance with the underlying intent of 10 CFR [Part] 50 Appendix G and provides operational limits established to prevent non-ductile failure of the reactor vessel. This Code case constitutes relaxation from the current requirements of 10 CFR [Part] 50 Appendix G. The alternate methodology allowed by the Code case is based on industry experience gained since the inception of the 10 CFR [Part] 50 Appendix G requirements and replaces some requirements that have now been determined to be excessively conservative. The more appropriate assumptions and provisions allowed by the Code case maintain a margin of safety that is consistent with the intent of 10 CFR [Part] 50 Appendix G. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Duke concludes that the proposed amendments present 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 amendment request involves no significant hazards consideration.

Attorney for licensee: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: December 12, 2002.

Description of amendment request: The proposed amendment would revise the Facility Operating License and Technical Specifications (TSs) to increase the licensed core thermal power level to 3114.4 megawatts (MWt), which is a 1.4% increase above the currently authorized power level of 3071.4 MWt. The proposed power uprate involves the improvement in the core power uncertainty allowance originally required for the emergency core cooling system (ECCS) evaluations performed in accordance with Appendix K, ''ECCS Evaluation Models,'' to Part 50 of Title 10 of the Code of Federal Regulations. In addition, changes would be made in TS Sections 1.1, 2.1, 2.3, 3.1, 3.4, 6.9, and the applicable TS Bases would be revised to account for the change in power level.

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

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

Response: No.

The proposed 1.4% increase in maximum core thermal power is based on the use of instrumentation that supports a reduction in the measurement uncertainty value assumed in certain safety analyses. The affected analyses now use an uncertainty value of 2% which was required by 10 CFR [Part] 50 Appendix K at the time that the plant was originally licensed. At that time, measurement of feedwater flowrate in the plant secondary side used differential pressure-type flow venturis. The plant secondary side thermal calorimetric is used to determine reactor thermal power. A June 2000 revision to 10 CFR [Part] 50 Appendix K permitted the use of lower uncertainty values in the affected analyses, if the reduced value can be justified. Entergy Nuclear Operations (ENO) has implemented the use of Caldon, Inc. Leading Edge Flowmeter (LEFM) technology to measure feedwater flowrate. The LEFM measures fluid velocity by measuring the transit time of ultrasonic pulses introduced into the fluid stream. The LEFM Check System implemented at Indian Point 2 has a demonstrated measurement accuracy of 0.6%. Based on this measurement accuracy, the licensed thermal power can be increased 1.4% by reducing the assumed uncertainty used in safety analyses with respect to core thermal power from 2.0% to 0.6%. This results in a net increase in licensed reactor core thermal power; from 3071.4 MWt to 3114.4 MWt. The LEFM and the flow venturi instrumentation are used to collect data and there is no automatic initiation function performed by this instrumentation. Use of the LEFM instrumentation is therefore not an accident initiator and does not increase the probability of occurrence of an existing analyzed accident. Also, the LEFM instrumentation and the venturi instrumentation do not mitigate accidents so that the consequences of previously analyzed accidents are not increased.

Analyses and evaluations associated with the proposed change to core thermal power have demonstrated that applicable acceptance criteria for plant systems, components, and analyses (including the Final Safety Analysis Report [FSAR] Chapter 14 safety analyses) will continue to be met for the proposed 1.4% increase in licensed core thermal power for Indian Point 2. The subject increase in core thermal power will not result in conditions that could adversely affect the integrity (material, design, and construction standards) or the operational performance of any potentially affected system, component or analysis. Therefore, the probability of an accident previously evaluated is not affected by this change. The subject increase in core thermal power will not adversely affect the ability of any safety-related system to meet its intended safety function. Further, the radiological dose evaluations in support of this power uprate effort show that the current FSAR Chapter 14 radiological analyses are unaffected, and that the current dose analyses of record bound plant operation with the subject increase in licensed core thermal power level.

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

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

Response: No.

The proposed license amendment increases the maximum allowed core thermal power through the use of feedwater flow instrumentation that supports a reduction in the measurement uncertainty assumed in certain safety analyses. The LEFM Check System instrumentation has greater measurement accuracy than the differential pressure-type flow venturi instrumentation that was originally used so that the measurement uncertainty assumed in certain analyses can be correspondingly reduced. Both the venturi and LEFM flow instrumentation provide data that is used by plant operators to monitor the thermal output of the plant. The instrumentation does not perform an automatic actuation function and there are no output signals to plant safety systems or control systems. Therefore, instrumentation malfunction or failure does not introduce new accident scenarios or equipment failure mechanisms. Operation, maintenance, or failure of either instrumentation system does not have an adverse effect on safety-related systems or any structures, systems, and components required for transient or accident mitigation.

Operating the plant at a new maximum core thermal power of 3114.4 MWt, which is 1.4% greater than the current maximum of 3071.4 MWt, is bounded by existing or updated analyses which demonstrate that established limits and acceptance criteria continue to be met. Operating at the new power level does not create new or different accident initiators and existing credible malfunctions are bounded by existing or updated analyses or evaluations.

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

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

Response: No.

The evaluations and analyses associated with the proposed increase in maximum core thermal power demonstrate that applicable acceptance criteria will continue to be met. The existing licensed maximum core thermal power level incorporates a 2% measurement uncertainty for the analysis of loss-of-coolant-accidents as originally required by Appendix K of 10 CFR [Part] 50. The regulations have subsequently been revised to allow the option of justifying smaller measurement uncertainties by using more accurate instrumentation to calculate reactor thermal power. Certain analyses that already assume a bounding core power level because of the 2% measurement uncertainty are not changed as a result of the proposed increase in core thermal power. Use of the LEFM instrumentation with improved measurement accuracy supports the use of a smaller measurement uncertainty assumption in the safety analyses. Other analyses were updated or evaluations were performed to demonstrate that nuclear steam supply and balance-of-plant systems and components will continue to perform, under normal and credible transient conditions, within established limits.

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

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

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

NRC Section Chief: Richard J. Laufer.

Exelon Generation Company, LLC, Docket No. 50–353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Date of amendment request: November 21, 2002.

Description of amendment request: Exelon Generation Company, LLC, the licensee, is proposing a change to the Limerick Generating Station (LGS), Unit 2, 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) due to the cycle-specific analysis performed by Global Nuclear Fuel for LGS, Unit 2, Cycle 8.

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

The derivation of the cycle specific Safety Limit Minimum Critical Power Ratios (SLMCPRs) for incorporation into the Technical Specifications (TS), and their use to determine cycle specific thermal limits, has been performed using the methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–14–US, June, 2000, which incorporates Amendment 25. Amendment 25 was approved by the NRC [Nuclear Regulatory Commission] in a March 11, 1999 safety evaluation report.

The basis of the SLMCPR calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling. The GE–14 fuel is in compliance with Amendment 22 to ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–14–US, June, 2000, which provides the fuel licensing acceptance criteria. The probability of fuel damage will not be increased as a result of this change. Therefore, the proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The SLMCPR is a TS numerical value, calculated to ensure that transition boiling does not occur in 99.9% of all fuel rods in the core if the limit is not violated. The new SLMCPRs are calculated using NRC approved methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–14–US, June, 2000, which incorporates Amendment 25. Additionally, the GE–14 fuel is in compliance with Amendment 22 to ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–14–US, June 2000, which provides the fuel licensing acceptance criteria. The SLMCPR is not an accident initiator, and its revision will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

There is no significant reduction in the margin of safety previously approved by the NRC as a result of the proposed change to the SLMCPRs, which includes the use of GE–14 fuel. The new SLMCPRs are calculated using methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–14–US, June, 2000, which incorporates Amendment 25. The SLMCPRs ensure that greater than 99.9% of all fuel rods in the core will avoid 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 the margin of safety previously approved by the NRC.

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, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: James W. Andersen.

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 26, 2002.

Description of amendment request: The proposed amendments revise Technical Specification (TS) 3.1.3.1, Control Rod Operability,'' by adding required actions for scram discharge volume (SDV) vent and drain valves to align with those in NUREG–1433, ''Standard Technical Specification, General Electric Plants, BWR/4,'' Revision 2. Additionally, modifications are proposed to change TS 3.6.3, ''Primary Containment Isolation Valves,'' to clarify the relationship between TS 3.1.3.1 and TS 3.6.3 regarding SDV vent and drain valve.

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

The scram discharge volume (SDV) and control rod drive (CRD) system, including the associated SDV vent and drain isolation valves, are not initiators to any accident sequence analyzed in the Updated Final Safety Analysis Report (UFSAR). Operation in accordance with the proposed Technical Specification (TS) ensures that the SDV and control rods are capable of performing their function as described in the UFSAR; therefore, the mitigative functions supported by the SDV and control rods will continue to provide the protection assumed by the analysis. The addition of specific TS actions to be taken for inoperable SDV vent or drain isolation valves will not challenge the ability of the SDV and control rods to perform their design function. Appropriate monitoring and maintenance, consistent with industry standards, will continue to be performed. In addition, the CRD system including the SDV isolation valves is within the scope of 10 CFR 50.65, ''Requirements for monitoring the effectiveness of maintenance at nuclear power plants,'' which will ensure the control of maintenance activities associated with the CRD system and SDV isolation valves.

Under the proposed TS changes, the SDV vent and drain lines may be unisolated under administrative control. This allows any accumulated water in the line to be drained, to preclude a reactor scram on SDV high level. This is acceptable since the administrative controls ensure the valve can be closed quickly, by a dedicated operator, if a scram occurs with the valve open. The 8-hour allowable outage time to isolate the line is based on the low probability of a scram occurring while the line is not isolated and unlikelihood of significant CRD seal leakage.

The proposed changes do not involve any physical change to structures, systems, or components (SSCs) and do not alter the method of operation or control of SSCs. The current assumptions in the safety analysis regarding accident initiators and mitigation of accidents are unaffected by these proposed changes. No additional failure modes or mechanisms are being introduced and the likelihood of previously analyzed failures remains unchanged.

The integrity of fission product barriers, plant configuration, and operating procedures as described in the UFSAR will not be affected by these proposed changes. Therefore, the consequences of previously analyzed accidents will not increase because of these proposed changes.

Based on the above discussion, the proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not involve a physical alteration of the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There are no setpoints, at which protective or mitigative actions are initiated, affected by these proposed changes. These proposed changes will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. Any alteration in procedures will continue to ensure that the plant remains within analyzed limits, and no change is required to the procedures relied upon to respond to an off-normal event as described in the UFSAR. As such, no new failure modes are being introduced. The changes do not alter assumptions made in the safety analysis and licensing basis.

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

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

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed changes are acceptable because the operability of the SDV and SDV isolation valves is unaffected, there is no detrimental impact on any equipment design parameter, and the plant will still be required to operate within assumed conditions. Operation in accordance with the proposed TS ensures that the SDV and control rods are capable of performing their functions as described in the UFSAR. Therefore, the support of the SDV and control rods in the plant response to analyzed events will continue to provide the margins of safety assumed by the analysis. The additions to TS for inoperable SDV vent and drain isolation valves will not challenge the ability of the SDV or control rods to perform their design function. Appropriate monitoring and maintenance, consistent with industry standards, will continue to be performed. In addition, CRD system, including the SDV vent and drain isolation valves, are within the scope of 10 CFR 50.65, ''Requirements for monitoring the effectiveness of maintenance at nuclear power plants,'' which will ensure the control of maintenance activities associated with the CRD system. This provides sufficient management control of the requirements that assure the control rods and CRD system are maintained in a highly reliable condition. Although there is an increase in allowable outage time, this increase was evaluated and determined not to be a significant reduction in a margin of safety.

The proposed TS Actions for inoperable SDV vent and drain isolation valves are reasonable and consistent with approved standards, guidance and regulations.

Based on the above discussion, the proposed TS 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: Mr. Edward Cullen, Vice President & General Counsel, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: James W. Andersen.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of amendment request: June 4, 2002.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) Surveillance Requirement (SR) 4.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period would be extended from ''* * * up to 24 hours to permit completion of the surveillance when the allowable (equipment inoperability) outage time limits of the ACTION requirements are less than 24 hours'' to ''* * * up to 24 hours or up to the limit of the specified frequency, whichever is greater.'' In addition, the following requirement would be added to SR 4.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours, and the risk impact shall be managed.'' The proposed amendment is consistent with TS Task Force traveler TSTF–358, which has been approved by the Nuclear Regulatory Commission (NRC). The TS Bases will be revised under the licensee's existing TS Bases control program to be consistent with the bases for TSTF–358.

Basis for proposed no significant hazards consideration determination: The NRC staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on September 28, 2001 (66 FR 49714). The licensee reviewed the model NSHC presented in the **Federal Register** and concluded that it is applicable to Davis-Besse. The model NSHC determination was incorporated by reference into its application dated June 4, 2002, to satisfy the requirements of 10 CFR 50.91(a), and is presented below:

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

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

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

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

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

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

Based upon the reasoning presented above and the previous discussion of the amendment request, the 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: December 9, 2002.

Description of amendment request: The proposed amendment utilizes the Alternate Source Term radiological calculations to update the design basis analysis in the Updated Safety Analysis Report for the Fuel Handling Accident. Regulatory Guide 1.183, ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,'' was utilized in the development of the proposed amendment.

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

The proposed amendment involves implementation of the Alternative Source Term for the Fuel Handling Accident at the Perry Nuclear Power Plant (PNPP). There are no physical design modifications to the plant associated with the proposed amendment. The revised calculations do not impact the initiators of a Fuel Handling Accident in any way. They also do not impact the initiators for any other design basis events. Therefore, because design basis accident initiators are not being altered by adoption of the Alternative Source Term analyses, the probability of an accident previously evaluated is not affected.

With respect to consequences, the only previously evaluated accident that could be affected is the Fuel Handling Accident. The Alternative Source Term is an input to calculations used to evaluate the consequences of an accident, and does not by itself affect the plant response, or the actual pathway of the radiation released from the fuel. It does however, better represent the physical characteristics of the release, so that appropriate mitigation techniques may be applied. For the Fuel Handling Accident, the AST analyses demonstrate acceptable doses, within regulatory limits, after 24 hours of radiological decay, without credit for Containment/Fuel Handling Building integrity, filtration system operability, or Control Room automatic isolation. Therefore, the consequences of an accident previously evaluated are not significantly increased.

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

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

The proposed amendment does not involve a physical alteration of the plant (no new or different type of equipment will be installed and there are no physical modifications to existing equipment associated with the proposed changes). Also, no changes are proposed to the methods governing plant/system operation during handling of recently irradiated fuel, so no new initiators or precursors of a new or different kind of accident are created. New equipment or personnel failure modes that might initiate a new type of accident are not created as a result of the proposed amendment.

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

3. This proposed amendment does not involve a significant reduction in a margin of safety.

The proposed amendment is associated with the implementation of a new licensing basis for PNPP Fuel Handling Accidents. Approval of the change from the original source term to a new source term taken from Regulatory Guide 1.183 is being requested. The results of the accident analyses, revised in support of the proposed license amendment, are subject to revised acceptance criteria. The analyses have been performed using conservative methodologies, as specified in Regulatory Guide 1.183. Safety margins have been evaluated and analytical conservatism has been utilized to ensure that the analyses adequately bound the postulated limiting event scenario. The dose consequences of the limiting Fuel Handling Accident remains within the acceptance criteria presented in 10 CFR 50.67, ''Accident Source Term,'' and Regulatory Guide 1.183.

The proposed changes continue to ensure that the doses at the exclusion area and low population zone boundaries, as well as the Control Room, are within corresponding regulatory limits. For the Fuel Handling Accident, Regulatory Guide 1.183 conservatively sets the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) limits below the 10 CFR 50.67 limit, and sets the Control Room limit consistent with 10 CFR 50.67.

Since the proposed amendment continues to ensure the doses at the EAB, LPZ and Control Room are within corresponding regulatory limits, the proposed license amendment does not involve a significant reduction in a margin of safety.

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

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

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

NRC Section Chief: Anthony J. Mendiola.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of amendment request: October 23, 2002.

Description of amendment request: The proposed amendment would revise Crystal River Unit 3 Improved Technical Specifications (ITS) 4.2.1, ''Fuel Assemblies,'' and ITS 4.2.2, ''Control Rods,'' to permit the use of Framatome ANP M5 advanced alloy for fuel rod cladding and fuel assembly structural components.

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:

Florida Power Corporation (FPC) has evaluated the proposed License Amendment Request (LAR), which consists of the identified Technical Specification changes and exemption requests, against the criteria of 10 CFR 50.92(c). The Technical Specification changes are categorized as follows:

1. Modification of Section 4.2.1, DESIGN FEATURES, Fuel Assemblies, and to include the M5 advanced alloy for fuel rod cladding and fuel assembly structural material[.]

2. Removal of design information such as maximum fuel enrichment, nominal active fuel length, maximum individual rod weight, and details of Control Rod content. Adopting the wording from the Standard ITS.

3. Addition to ITS 4.2.1 of the following sentence: ''A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.'' Crystal River Unit 3 does not intend to load lead test assemblies in the upcoming fuel cycle (Cycle 14). This sentence is being added for consistency with NUREG 1430, Revision 2.

FPC has concluded that this proposed LAR does not involve a significant hazards consideration. The following is a discussion of how each of the criteria is satisfied.

(1) [Does not] [i]nvolve a significant increase in the probability or consequences of an accident previously evaluated.

M5 advanced alloy: Topical reports BAW–10227P–A, ''Evaluation of Advanced Cladding and Structural Material (M5) in PWR [Pressurized Water Reactor] Reactor Fuel,'' February 2000 and BAW–10179P–A, Revision 4, ''Safety Criteria and Methodology for Acceptable Cycle Reload Analyses,'' March 2001 provide the licensing basis for the Framatome ANP (FRA–ANP) advanced cladding and structural material, designated M5. The M5 material can be used for fuel rod cladding, as well as for fuel assembly spacer grids, fuel rod end plugs, and fuel assembly guide and instrument tubes. By letter dated August 2, 2001 (Reference 4), the NRC approved BAW–10179P–A, Revision 4, for referencing in license applications. BAW–10179P–A, Revision 4 incorporates BAW–10227P–A. The M5 material was shown in these documents to have equivalent or superior properties to the current Zircaloy-4 material. The cladding itself is not an accident initiator and does not affect accident probability. The M5 cladding has been shown to meet all 10 CFR 50.46 design criteria and, therefore, will not increase the consequences of an accident.

Removal of design parameters of maximum fuel enrichment, active fuel length, rod weight and Control Rod content: This change moves design features from Improved Technical Specifications (ITS) to the Final Safety Analysis Report (FSAR) and other design documents and analyses. The Framatome ANP enhanced fuel design will involve increased rod weight and active fuel length. The approved Framatome ANP topical report, BAW–10179P–A, ''Safety Criteria and Methodology for Acceptable Cycle Reload Analyses,'' will continue to be used to ensure that the required safety limits for the fuel are satisfied. Therefore, the relocation of design information does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Addition of a limited number of lead test assemblies: This change is administrative in nature and is proposed for consistency with the ITS standard. Crystal River Unit 3 does not intend to load lead test assemblies in the upcoming fuel cycle. When lead test assemblies are to be loaded, the approved Framatome ANP topical report BAW–10179P–A will be used to ensure that all applicable limits of the safety analysis are met and that the lead test assemblies are placed in nonlimiting core locations. Applicable mixed core penalties and core operating limits will be developed and applied. Therefore, use of lead test assemblies will not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) [Does not] [c]reate the possibility of a new or different kind of accident from any accident previously evaluated.

M5 advanced alloy: Topical report BAW–10227P–A demonstrated that the material properties of the M5 alloy are not significantly different from those of Zircaloy-4. Therefore, M5 fuel rod cladding and fuel assembly structural components will perform similarly to those fabricated from Zircaloy-4, thus precluding the possibility of the fuel becoming an accident initiator and causing a new or different type of accident.

Removal of design parameters of maximum fuel enrichment, active fuel length, rod weight and Control Rod content: This change moves design features from ITS to the FSAR and other design documents and analyses or adds consistency with the standard ITS. The location of this information does not create the possibility of a new or different kind of accident from any accident previously evaluated. The approved FRA–ANP topical report, BAW–10179P–A will continue to be used to ensure that the required safety limits are satisfied. Therefore, these changes do not involve the possibility of a new or different kind of accident from any accident previously evaluated.

Addition of a limited number of lead test assemblies: This change is administrative in nature and it is proposed for consistency with the ITS standard. Crystal River Unit 3 does not intend to load lead test assemblies in the upcoming fuel cycle. When lead test assemblies are to be loaded, they will be designed and manufactured to ensure compatibility with the co-resident fuel assemblies, core internal structures, and fuel handling and storage equipment. The approved Framatome ANP topical report BAW–10179P–A will be used to ensure that the lead test assemblies meet all applicable limits of the safety analysis and that the lead test assemblies are placed in non-limiting core locations. Applicable mixed core penalties and core operating limits will be developed and applied. Therefore, use of lead test assemblies will not involve the possibility of a new or different kind of accident from any previously evaluated.

(3) [Does not] [i]nvolve a significant reduction in a margin of safety.

M5 advanced alloy: The proposed changes will not involve a significant reduction in the margin of safety because it has been demonstrated that the material properties of the M5 alloy are not significantly different from those of Zircaloy-4. The M5 alloy is expected to perform similarly or better [than] Zircaloy-4 for all normal operating and accident scenarios, including both non-LOCA [loss-of-coolant accident] and LOCA scenarios. For LOCA scenarios, where the slight differences in M5 material properties relative to Zircaloy-4 could have some impact on the overall accident scenario, plant-specific LOCA analyses will be performed prior to the use of fuel assemblies with fuel rods or fuel assembly components containing M5. These LOCA analyses, required by ITS 5.6.2.18, ''Core Operating Limits Report (COLR),'' will demonstrate that all applicable margins of safety will be maintained by the use of the M5 alloy.

Removal of design parameters of maximum fuel enrichment, active fuel length, rod weight and Control Rod content: Approved methodologies will be used in the cycle-specific safety analysis to evaluate the use of the M5 advanced alloy, and account for various assembly differences (various rod weights and active fuel lengths). The location of the design information does not affect the margin of safety.

Addition of a limited number of lead test assemblies: This change is administrative in nature and is proposed for consistency with the ITS standard. Crystal River Unit 3 does not intend to load lead test assemblies in the upcoming fuel cycle. When lead test assemblies are to be loaded, the approved Framatome ANP topical report BAW–10179P–A will be used to ensure that all applicable limits of the safety analysis are met and that the lead test assemblies are placed in nonlimiting core locations. Applicable mixed core penalties and core operating limits will be developed and applied. There will be no significant reduction in the margin of safety when a limited number of lead test assemblies are utilized.

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

Attorney for licensee: R. Alexander Glenn, Associate General Counsel (MAC–BT15A), Florida Power Corporation, P.O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Section Chief: Allen G. Howe.

Florida Power and Light Company, Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of amendment request: November 25, 2002.

Description of amendment request: The proposed license amendment would modify plant Technical Specifications (TSs) and the associated spent fuel pool (SFP) criticality analyses to eliminate credit for the Boraflex™ neutron absorber in SFP fuel storage racks and credit specific rules to control fuel assembly positioning in the SFP racks. TS 3.9.11 is revised to add a Limiting Condition for Operation for the SFP soluble boron concentration and require periodic surveillance of this parameter. This submittal provides justification for removing the description of the poison material in the spent fuel racks from Section 5 of the Unit 1 TSs, that was requested to be added by the licensee's cask pit spent fuel storage rack submittal dated October 23, 2002. In addition, a new SFP dilution analysis was performed that supports the criticality analysis requirement for a minimum soluble boron concentration.

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) Would operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed amendment to eliminate reliance on Boraflex™ and to credit SFP soluble boron for reactivity control in the spent fuel pool storage racks was evaluated for impact on the following previously evaluated events:

- A fuel handling accident (FHA)
- A fuel mispositioning event
- A cask drop accident
- A loss of spent fuel pool cooling

The proposed amendment does not modify the facility. A new criticality analysis credits existing soluble boron in the SFP water and specific fuel positioning rules for reactivity control, without requiring any physical changes to the fuel storage racks. The amendment does not change any rack module location or any module's designation as Region 1 or Region 2 storage. There is no significant increase in the probability of a fuel handling accident in the SFP that is caused by crediting soluble boron and new fuel positioning rules, rather than Boraflex™, for reactivity control. The probability of a fuel handling accident is a function of the equipment design and procedures used when handling irradiated fuel. Neither of these features is affected when soluble boron, instead of Boraflex™, is credited for reactivity control in the SFP.

There is no increase in the probability of an accidental fuel assembly mispositioning when crediting the presence of soluble boron in fuel pool water for reactivity control. Fuel assembly selection and manipulation will continue to be controlled by approved fuel handling procedures; these procedures require the identification of a verified target location prior to grappling the assembly. Fuel placement will be in accordance with the revised TS.

There is no increase in the consequences of either an FHA or an accidental mispositioning of a fuel assembly into the SFP racks. Consequences of a FHA are not increased because the proposed amendment does not change the fuel fission product inventory, local meteorological conditions, or the fission product partition factor provided by fuel pool water. The consequences of an accidental misload are not increased because the criticality analysis demonstrates that the fuel array will remain sub-critical, even if the pool contains a boron concentration below the minimum level required by Technical Specifications. The TS will ensure that an adequate SFP soluble boron concentration is maintained for all conditions.

The proposed fuel positioning rules do not cause the total radionuclide inventory present in the spent fuel pool to increase, or alter the type or mass of casks that may be placed in the fuel pool, or alter any facet of operation of the spent fuel cask crane. No characteristics of the existing spent fuel cask drop analysis for Unit 1 are affected by the proposed fuel positioning rules or by credit for soluble boron. Therefore, there is no increase in either the probability or the consequences of a cask drop accident caused by this change.

The proposed change does not increase either the probability or the consequences of a loss of normal SFP cooling. The proposed fuel positioning rules do not require any interaction with the fuel pool cooling system. Credit for a portion of the existing soluble boron concentration does not change its interaction with the fuel pool cooling system. The ability to detect and mitigate a loss of SFP cooling event is unchanged, and the revised criticality analysis considered the effects of boiling in the SFP and found them acceptable.

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

(2) Would operation of the facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed change does not modify the physical plant, nuclear fuel, or the design function and operation of the spent fuel pool storage racks at St. Lucie Unit 1. A TS controlled minimum concentration of soluble boron has always been required in the St. Lucie Unit 1 spent fuel pool; as such, the possibility of an inadvertent fuel pool dilution event has always existed. However, the spent fuel pool dilution analysis that accompanies this submittal demonstrates that no credible dilution event could increase fuel pool reactivity such that the effective neutron multiplication factor (k_{eff}) exceeds 0.95. Therefore, implementation of credit for soluble boron to control reactivity in the SFP will not create the possibility of a new or different type of criticality accident.

The limiting fuel assembly mispositioning event does not represent a new or different type of accident. The mispositioning of a fuel assembly within the fuel storage racks has always been possible. The locations of SFP rack modules and the specific modules assigned to each storage region remain unchanged; analysis results show that the storage racks remain subcritical, with substantial margin, following a worst case fuel misloading event. Therefore, a fuel assembly misload event that involves new fuel storage arrangements required by the criticality analysis does not result in a new or different type of criticality accident.

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

(3) Would operation of the facility in accordance with the proposed amendment involve a significant reduction in a margin of safety?

No. The revised fuel positioning requirements proposed by this license amendment provide sufficient safety margin to ensure that the spent fuel pool storage racks will always remain subcritical. To comply with the requirements of 10 CFR 50.68 when crediting soluble boron, the current TS reactivity limit for the fuel storage racks (*i.e.,* k_{eff} less than or equal to 0.95 when flooded with unborated water) will be replaced with two separate limits (k_{eff} less than 1.0 when flooded with unborated water, and k_{eff} less than or equal to 0.95 when flooded with water containing 500 ppm boron).

The proposed amendment maintains the 0.95 reactivity limit by a combination of restrictions on fuel characteristics and fuel positioning, storage cell geometry and by crediting a portion of the soluble boron in the SFP, rather than by crediting Boraflex.

The proposed license amendment does not reduce the margin of safety provided by the soluble boron normally present in fuel pool water; the TS minimum permissible boron concentration is not decreased. The TS minimum required value of 1720 ppm is substantially greater than the 500 ppm value required by the updated criticality analysis to assure k_{eff} remains = 0.95 for non-accident conditions; it is also substantially greater than the soluble boron concentration necessary to compensate at a 95% probability, with a 95 percent confidence for the limiting postulated reactivity anomaly in the fuel pool storage racks.

No credible dilution of the fuel pool can result in an SFP soluble boron concentration less than the minimum value required by the criticality analysis. Therefore, an inadvertent dilution event can not challenge safety margins.

Based on these evaluations and the supporting analyses, operating the facility with the proposed amendment does not involve in a significant reduction in any margin of safety.

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

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

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: April 22, 2002, as supplemented on December 5, 2002.

Description of amendment request: The proposed amendment would allow removal of the upper half of the SNEF containment vessel and make 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.

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

GPU Nuclear has determined that Technical Specification Change Request No. 62 involves no significant hazard consideration as defined in 10 CFR 50.92.

1. The proposed changes to the SNEC Technical Specifications do not involve a significant increase in the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously analyzed in the safety analysis report.

As described in the change to delete Technical Specification 1.1.2, radiation levels inside the Containment Vessel will be below that necessary to maintain the Containment Vessel as an Exclusion Area. Further as required by modified Technical Specification 2.1.1 ventilation controls will be established to monitor and control any potential releases of airborne radioactivity during activities involving removal of the upper dome. Finally an analysis has been performed to determine the dose to a maximally exposed individual due to an accidental release while cutting the Containment Vessel. In developing a source term for the event it was assumed that following the concrete removal process the interior surfaces of the upper Containment Vessel dome was homogeneously coated with concrete dust. NUREG 1507 ''Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions'' describes an experiment to determine the attenuation effects due to dusty conditions. The maximum dust loading presented was 9.99 mg/cm^2 for soil. This value was converted to concrete dust by comparing the relative densities of the material (1.5 g/cm^3 for soil and 2.3 g/cm^3 for concrete) or 15.3 mg/cm^2. This amount of dust coating the internal surfaces of the Containment Vessel dome (9.05E6 cm^2) results in 299 pounds of dust being left in the Containment Vessel.

Table 1 provides the mix of isotopes remaining at the SNEC Facility based on the most recent survey results and isotope decay. During the removal operation a resuspension factor of 1.9E–2/m (as described in NUREG/CR 0130 ''Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station'', Volume 2, page J–27) was selected to represent the amount of concrete dust going airborne. This parameter is about one order of magnitude larger than that used in any other accident analyses described in the NUREG. This entire volume of dust was assumed to be released, unfiltered, directly to the environment.

An accident dispersion factor (c/Q) of 3.41E–3 sec/m^3, was also selected as it is the highest, thus most conservative, value used in the SNEC Facility Offsite Dose Calculation Manual (ODCM). Additionally composite dose conversion factors were selected from

Table 5–1 of EPA 400–R–92–001 ''Manual of Protective Action Guides and Protective Guides for Nuclear Incidents'' (US EPA, May 1992).

Based on the above a calculated dose of 3.23E–4 mrem to the maximally exposed individual represents a conservative estimate for an accidental release. For comparison Section 3.1 of the SNEC Facility USAR estimated the dose from an unfiltered release due to a material handling event of 1.5 mrem to the maximally exposed individual.

Thus this proposed change does not involve a significant increase in the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously analyzed in the SNEC Facility USAR.

For the portions of the amendment that would make 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, these changes are administrative in nature. As such they have no effect on the probability of occurrence or consequences of an accident or malfunction of equipment important to safety.

2. The proposed changes to the SNEC Technical Specifications will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the safety analysis report.

As described in the response to item 1 above, the limiting accidental release during segmentation of the Containment Vessel dome involves the direct release of radioactive material to the environment. This event is similar to both a material handling event as described in Section 3.1 of the SNEC Facility USAR, and loss of engineering controls during segmentation as described in Section 3.4 of the SNEC Facility USAR. Thus the possibility of a new accident is not created.

For the portions of the amendment that would make 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, these changes are administrative in nature. As such they have no effect on the possibility of an accident or malfunction of a different type.

3. The changes will not involve a significant reduction in the margin of safety as defined in the basis for any technical specification for SNEC. The SNEC Facility Technical Specifications do not contain a defined margin of safety. However the implied margin of safety is to protect members of the public from exposure to radioactive material.

At the point in time that these Technical Specifications would take affect general radiation levels in the SNEC Facility Containment Vessel would be such that the Containment Vessel could be opened for unrestricted use as defined in 10 CFR 20.1301. Additionally the dose to a maximally exposed individual from an accidental release during removal of the Containment Vessel dome is several orders of magnitude below that from the limiting accidents defined in the SNEC Facility USAR. Thus the margin of safety is not reduced.

For the portions of the amendment that would make 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, these changes are administrative in nature. As such they have no effect on the margin of safety as defined in the basis for any technical specification for SNEC.

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

Attorney for the Licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Program Director: William D. Beckner.

TABLE 1.—MAXIMUM EXPOSED INDIVIDUAL DOSE FROM CUTTING THE CV

Isotope	CV concrete activity (Ci) per table 4.13 SNEC char. report	Fraction remaining as dust (uCi)	CV wall area concetration (uCi/m) [2]	CV air concetration (uCi/m) [3]	Instantaneous release rate (uCi/sec) [4]	Concentration (uCi/cm) [3]	DCF [7]	Offsite dose (mrem)
Am–241	8.24e–05	4.68e–03	5.17e–06	9.83e–08	2.93e–04	9.99e–13	1.47e+05	1.47e–04
Co–60	4.60e–02	2.61e+00	2.89e–03	5.49e–05	1.63e–01	5.57e–10	7.50e+01	4.18e–05
Cs–137	2.38e–01	1.35e+01	1.49e–02	2.84e–04	8.46e–01	2.88e–09	1.14e+01	3.28e–05
C–14	5.74e–03	3.26e–01	3.60e–04	6.84e–06	2.04e–02	6.96e–11	6.94e–01	4.83e–08
Eu–152	1.42e–03	8.07e–02	8.91e–05	1.69e–06	5.05e–03	1.72e–11	7.50e+01	1.29e–06
H–3	1.29e–01	7.33e+00	8.10e–03	1.54e–04	4.58e–01	1.56e–09	2.14e–02	3.34e–08
Ni–63	3.93e–02	2.23e+00	2.47e–03	4.69e–05	1.40e–01	4.76e–10	2.11e+00	1.01e–06
Pu–239	5.24e–05	2.98e–03	3.29e–06	6.25e–08	1.86e–04	6.35e–13	1.44e+05	9.17e–05
Pu–241	1.84e–04	1.05e–02	1.15e–05	2.19e–07	6.54e–04	2.23e–12	2.75e+03	6.13e–06
Sr–90	1.59e–04	9.03e–03	9.98e–06	1.90e–07	5.65e–04	1.93e–12	4.44e+02	8.56e–07
Total	4.60e–01	2.61e+01			1.63e+00			2.70e+05

[1] Fraction remaining determined by: (299 lbs dust/5.26E6 lbs total concrete in CV) × 1E6 uCi/Ci × CV concrete activity.
[2] Area concentration determined by dividing dust fraction remaining by 9.05E2 m^2 (surface of CV shell being removed).
[3] Air concentration determined by multiplying CV wall area activity by 1.9E–2/m (NUREG 0130 resuspension factor for dust sweeping).
[4] Calculated by multiplying CV air specific activity by CV volume (2.98E3 m^3) instantaneously released in one second.
[5] Maximum atmospheric dispersion factor (X/Q) is 3.41E–3 sec/m^3 at the site boundary (200 meters) and in Sector N per SNEC ODCM Revision 5.
[6] Calculated by multiplying X/Q × activity released in uCi/sec × 1e–6 m^3/cm^3.
[7] Per EPA 400–R–92–001, Table 5–1.

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

Date of amendment request: July 10, 2002.

Description of amendment request: This proposed amendment would replace the fire protection (FP) requirements contained in Facility Operating License (FOL) Section 2.C.(4) with the standard fire protection FOL condition recommended by Generic Letter 86–10, Section F, adapted to Cooper Nuclear Station (CNS).

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

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

The proposed change would revise the CNS Operating License condition concerning the FP program and its change process. It does not alter the FP requirements in the FHA [fire hazard analysis] or in the USAR [updated safety analysis report] including the assumptions underlying them. Neither does it alter SSCs [structures, systems or components] relied on by analyses to mitigate accidents or special events. Since it does not change any of the FP requirements or analyses, this proposed amendment does not introduce a new initiator for any of the accidents analyzed in the CNS USAR or considered therein. Because it does not specifically change any FP requirements or mitigating SSCs, this proposed amendment does not introduce a new mechanism for degrading the mitigating features considered for the accidents analyzed. By introducing no new accident initiators and no new mechanisms for degradation of mitigating features, no significant increase in the probability or consequences of an accident previously evaluated is involved in the proposed change. Therefore, the proposed change does not result in a significant increase in radiological doses for any Design Basis Accident and does not result in a significant increase in the types or amounts of any effluents that may be released off-site.

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

The proposed amendment does not physically change the fit, form, or function of any SSC credited in the accident analyses or in the FHA, Technical Requirements Manual (TRM), or the USAR. The proposed change does not alter assumptions or requirements used in the FHA, TRM, or USAR, nor does it affect the CNS Fire Protection program. It does not, therefore, alter the FP program or affect the plant's ability to achieve and maintain safe shutdown in the event of a fire, and it does not result in a reduction in the level of fire protection of the facility. Because it does not change FP requirements, the FP program or fire-mitigating SSCs, this proposed change does not create the possibility of a new or different kind of accident from those previously evaluated for CNS.

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

The proposed amendment does not alter the design features of the approved FP plan. The proposed amendment does not alter administrative controls in the CNS Fire Protection program necessary to ensure required performance of physical barriers during anticipated operational occurrences and postulated accidents. The proposed change does not alter the NRC approved Fire Protection program as described in FP SER [safety evaluation report] dated May 23, 1979, SER Supplement 1 dated November 21, 1980, SER dated September 21, 1983, SER dated April 16, 1984, SER dated August 21, 1985, SER dated April 10, 1986, SER dated November 7, 1988, SER dated August 15, 1995. It does not affect the USAR, the TRM, the FHA or the commitments contained therein. It does not physically change the fit, form, or function of any SSC credited in the accident analyses or in these documents. Because it does not change the requirements, plan or mitigating SSCs, this proposed change does not involve a significant reduction in a margin of safety.

In summary, the proposed amendment does not involve a significant increase in the probability or consequences of an accident or creates the possibility of a new or different kind of accident or 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: Mr. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: November 22, 2002.

Description of amendment request: The proposed amendment would revise the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS), Section 4.6, ''Periodic Testing of Emergency Power System.'' This proposed amendment would allow KNPP to inspect the diesel generators (DGs) at least once per refueling frequency either while the plant is operating or during a refueling outage. Current TS requires an inspection during the refueling outage without exception. In addition, the proposed amendment would allow KNPP to make administrative changes to TS Section 4.6. The proposed change provides operational flexiblity in the schedule of maintenance activities.

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

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The DGs are accident mitigating equipment, not accident initiating equipment. Consequently, there will be no impact on any accident probabilities by the approval of the requested amendment.

The proposed change does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. Consequently, no analysis assumptions are violated and there are no adverse effects on the factors that contribute to off-site or on-site dose as the result of an accident.

The format, typographical, grammatical, and standardized naming convention changes in addition to the WORD conversion are administrative in nature and therefore have no impact on accident initiators or plant equipment.

Based on the above, the proposed administrative changes and permitting DG inspections to be performed during plant operation does not involve a significant increase in the probabilities or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

No new accident mechanisms would be created as a result of NRC approval of this amendment request since no changes are being made to the plant that would introduce any new accident mechanisms. Equipment would be operated in the same configurations with the exception of the mode in which the inspection is credited. The inspection will be performed within the current approved Technical Specification limiting condition for operation (LCO). This amendment request does not impact any plant systems that are accident initiators or adversely impact any accident mitigating systems.

The proposed administrative changes do not involve any modifications to the physical plant or operations. Administrative changes do not contribute to accident initiators nor do they produce a new accident scenario. Based on the above, implementation of the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in the margin of safety.

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include fuel cladding, the reactor coolant system, and the containment system. The proposed change to the inspection timing for the DGs do not affect the operability requirements for the DGs, as verification of such operability will continue to be performed as required. Continued verification of operability supports the capability of the DGs to perform their required function of providing emergency power to plant equipment that supports the fission product barriers. Consequently, the performance of these fission product barriers will not be impacted by implementation of this license amendment request and therefore does not involve a significant reduction in the margin of safety.

The administrative changes do not affect plant equipment or operation. 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 H. O'Neill, Jr., Esq., Shaw Pittman, Potts & Trowbridge, 2300 N. Street, NW., Washington, DC 20037–1128.

NRC Section Chief: L. Raghavan.

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: August 27, 2002.

Description of amendment requests: The proposed license amendments would revise the term ''minimum measured flow per loop'' to ''measured loop flow'' in the allowable value and nominal trip setpoint for the Reactor Coolant Flow-Low reactor trip function contained in Table 3.3.1–1, ''Reactor Trip System Instrumentation,'' of Technical Specification (TS) 3.3.1. In addition, the proposed amendments would allow for an alternate method for the measurement of reactor coolant system (RCS) total volumetric flow rate through measurement of the elbow tap differential pressures on the RCS primary cold legs. The use of elbow tap differential pressures normalized to Diablo Canyon Power Plant Cycle 1 and 2 precision flow calorimetrics would improve the accuracy of the RCS flow measurement through reduction of the effect of hot leg temperature streaming that is present in the current flow calorimetric method.

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

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

The proposed change revises the Technical Specification (TS) 3.3.1 Table 3.3.1–1 term ''minimum measured flow per loop'' to ''measured loop flow'' in the allowable value and nominal trip setpoint for the Reactor Coolant Flow-Low reactor trip function and allows an alternate method for the measurement of reactor coolant system (RCS) total flow to meet TS surveillance requirement (SR) SR 3.4.1.4 through measurement of the elbow tap differential pressures on the RCS primary cold legs.

The change will not increase the probability of an accident previously evaluated because adequate RCS flow will still be assured. The Reactor Coolant Flow-Low reactor trip function allowable value and nominal trip setpoint are accident mitigation functions and are not an accident initiator. The elbow tap method to measure RCS flow and the change to the flow definition associated with the Reactor Coolant Flow-Low reactor trip function do not involve a plant modification.

For the elbow tap method to measure RCS flow, sufficient margin exists to account for all reasonable instrument uncertainties and therefore the RCS flow will continue to be maintained at a value which is bounded by the design basis accident initial conditions. The change to the flow definition associated with the Reactor Coolant Flow-Low reactor trip function allowable value and nominal trip setpoint does not change a design basis accident initial condition or the conditions at the time of reactor trip during a design basis accident and therefore has no adverse effect on the design basis accidents which credit the Reactor Coolant Flow-Low reactor trip setpoint.

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

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

The proposed change to the flow definition associated with the Reactor Coolant Flow-Low reactor trip function allowable value and nominal trip setpoint and the proposed elbow tap method to measure RCS flow will not create the possibility of a new or different type of accident from any previously evaluated. There are no physical changes being made to the plant and there are no changes in operation of the plant that could introduce a new failure mode, creating an accident which has not been evaluated.

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.

The proposed change to the flow definition associated with the Reactor Coolant Flow-Low reactor trip function allowable value and nominal trip setpoint and the proposed elbow tap method to measure RCS flow will not reduce the margin of safety. For the proposed elbow tap flow method, sufficient margin exists to account for all reasonable instrument uncertainties and thus the RCS flow will continue to be maintained at a value which is bounded by the design basis accident initial conditions, and no adverse effect on the plant response to design basis accidents is created. The change in the flow definition associated with the Reactor Coolant Flow-Low reactor trip function allowable value and nominal trip setpoint does not change a design basis accident initial condition or the conditions at the time of reactor trip during a design basis accident, and therefore has no effect on the plant response to design basis accidents which credit the Reactor Coolant Flow-Low reactor trip setpoint. Since the change does not affect the response to design basis accidents, it does not result in a decrease in departure from nucleate boiling margin or reactor coolant system peak pressure margin for the design basis accidents.

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: Christopher J. Warner, 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: November 1, 2002.

Description of amendment requests: The proposed license amendments would revise Technical Specification (TS) 3.3.1, ''Reactor Trip System (RTS) Instrumentation,'' and TS 3.3.2, ''Engineered Safety Feature Actuation System (ESFAS) Instrumentation'' as follows: (1) Revise both the RTS and ESFAS instrumentation TS and TS Bases to change or clarify the allowances for bypassing and tripping tested channels with other channels inoperable; (2) remove Surveillance Requirement 3.3.1.10 from Function 16.b, ''Turbine Stop Valve Closure;'' (3) correct the nominal trip setpoint value for Function 16.b, ''Turbine Stop Valve Closure;'' (4) correct the allowable value for the Function 18.f, ''Turbine Impulse Chamber Pressure, P–13;'' and (5) remove and relocate the nonsafety-related turbine trip function from Function 5 of Table 3.3.2–1, ''Turbine Trip and Feedwater Isolation.'' This function will be relocated to other owner-controlled documents.

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

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

The proposed changes in the required action statements in the Limiting Conditions for Operation (LCOs) for the allowable surveillance testing configurations for both the reactor trip system (RTS) and engineered safety feature actuation system (ESFAS) instruments will not change the probability or consequences of an accident previously evaluated.

The proposed surveillance testing configuration changes only clarify available surveillance testing configurations and limitations on those configurations. The changes do not modify how the RTS and ESFAS functions respond to any accident condition. These surveillance testing configurations provide greater flexibility to prevent inadvertent actuation of these functions that could be a precursor for an accident.

Previous Diablo Canyon Power Plant (DCPP) submittals have been approved providing for the capability of surveillance testing in trip and/or in bypass. Surveillance testing in bypass is considered the preferred method for most Eagle 21 instruments. However, where testing by tripping a single channel without causing a function actuation is acceptable, that capability was also maintained.

Although some of the changes may appear to add new allowable surveillance testing configurations, all of the proposed configurations are based on the application of the intent behind the existing Technical Specification (TS) wording. The limitations on surveillance testing configurations provided by the proposed changes are to ensure that there are no spurious actuations and that during testing a valid signal will cause the associated functions to actuate as designed. None of these configurations place the associated function in a logic that has not been previously evaluated and approved.

The proposed elimination of the channel calibration for the turbine stop valve position switches will not change the probability or consequences of an accident previously evaluated since these switches are not subject to drift. These limit switches are installed with fixed limit setpoints that actuate based on valve position and they are not calibrated in the field. As a result, a channel calibration being performed on these switches provides no useful purpose other than to verify function similar to the remaining trip actuation device operational test (TADOT). As a result, performing only the TADOT provides all necessary assurances of operability.

The correction of the turbine stop valve closure nominal trip setpoint is administrative in nature and will not change the probability or consequences of an accident previously evaluated. This was an oversight in the Improved Technical Specification (ITS) review and conversion process. The proposed change only returns the setpoint to the previously evaluated value.

The proposed change to the allowable value for Function 18.f, ''Turbine Impulse Chamber Pressure, P–13,'' is administrative in nature and will not change the probability or consequences of an accident previously evaluated. The P–13 intended trip setpoint has always been maintained at 10 percent and remains unchanged. This modification is performed to provide consistency with current methodology and NUREG–1431, and does not affect the operation of the protective function.

The proposed removal and relocation of the turbine trip function from ESFAS Function 5 will not change the probability or consequences of an accident previously evaluated. The turbine trip function is nonsafety-related and is not credited in any design bases accident scenario. The proposed change only clarifies importance of the two trip functions. The proposed changes in this LAR [License Amendment Request] 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 in the required action statements in the LCOs for the allowable surveillance testing configurations for both the RTS and ESFAS instruments will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes only clarify previously available surveillance testing configurations and limitations on those configurations. These clarifications ensure maximum surveillance testing flexibility to prevent inadvertent actuation of these functions that could be a precursor for an accident. The changes do not modify any equipment, hardware or how the RTS and ESFAS functions respond to any accident condition.

The proposed elimination of the channel calibration for the turbine stop valve position switches will not create the possibility of a new or different kind of accident from any accident previously evaluated. This change does not modify any equipment, hardware or functions. The switches are installed with fixed limit setpoints that actuate based on valve position. The switches are not subject to drift and are not calibrated in the field. As a result, a channel calibration being performed on these switches provides no useful purpose other than to verify function similar to the required TADOT. As a result, performing only the TADOT provides equivalent assurances of operability.

The correction of the turbine stop valve closure nominal trip setpoint in Function 16.b, ''Turbine Stop Valve Closure,'' is administrative in nature and will not create the possibility of a new or different kind of accident from any accident previously evaluated. This was an oversight in the ITS review and conversion process. The proposed change does not modify any hardware or equipment, and only returns the setpoint to the previously evaluated value.

The proposed change to the allowable value for Function 18.f, ''Turbine Impulse Chamber Pressure, P–13,'' is administrative in nature and will not create the possibility of a new or different kind of accident from any accident previously evaluated. The P–13 intended (nominal) trip setpoint has always been maintained at 10 percent and remains unchanged. This change does not modify any equipment or hardware. This modification is performed to provide consistency with current methodology and NUREG–1431, and does not affect the operation of the protective function.

The proposed removal and relocation of the turbine trip function from ESFAS Function 5 will not create the possibility of a new or different kind of accident from any accident previously evaluated. The turbine trip function is nonsafety-related and is not credited in any design bases accident scenario. The proposed change only clarifies importance of the two trip functions.

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

The proposed changes in the required action statements in the LCOs for the allowable surveillance testing configurations for both the RTS and ESFAS instruments will not involve a significant reduction in a margin of safety. The proposed changes only clarify previously available surveillance testing configurations and limitations on those configurations. These clarifications ensure maximum surveillance testing flexibility to prevent inadvertent actuation of these functions that could be a precursor for an accident. The changes do not modify any equipment, hardware or how the RTS and ESFAS functions respond to any accident condition.

The proposed elimination of the channel calibration for the turbine stop valve position switches will not involve a significant reduction in a margin of safety. This change does not modify any equipment, hardware or functions. The switches are installed with fixed limit setpoints that actuate based on valve position. The switches are not subject to drift and are not calibrated in the field. As a result, a channel calibration being performed on these switches provides no useful purpose other than to verify function similar to the required TADOT. As a result, performing only the TADOT provides equivalent assurances of operability.

The correction of the turbine stop valve closure nominal trip setpoint in Function 16.b, is administrative in nature and will not involve a significant reduction in a margin of safety. This was an oversight in the ITS review and conversion process. The proposed change does not modify any hardware or equipment, and only returns the setpoint to the previously evaluated value.

The proposed change to the allowable value for Function 18.f, ''Turbine Impulse Chamber Pressure, P–13,'' is administrative in nature and will not involve a significant reduction in a margin of safety. The P–13 intended (nominal) trip setpoint has always been maintained at 10 percent and remains unchallenged. This change does not modify any equipment or hardware. This modification is performed to provide consistency with current methodology and NUREG–1431, and does not affect the operation of the protective function.

The proposed removal and relocation of the turbine trip function from ESFAS Function 5 does not involve a significant reduction in a margin of safety. The turbine trip function is nonsafety-related and is not credited in any design bases accident scenario. The proposed change only clarifies importance of the two trip functions.

None of the proposed changes affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions.

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: Christopher J. Warner, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Section Chief: Stephen Dembek.

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

Date of amendments request: December 9, 2002.

Description of amendments request: The proposed amendments would revise Technical Specification 3.7.5, ''Auxiliary Feedwater System,'' Surveillance Requirement (SR) 3.7.5.2 for San Onofre Nuclear Generating Station, Units 2 and 3. Specifically, the proposed change would change wording of the Frequency of SR 3.7.5.2 from ''31 days on a Staggered Test Basis'' to ''In accordance with the Inservice Testing Program.'' Such inservice tests confirm component operability, trend performance, and detect incipient failures by indicating abnormal performance. This change is requested to implement recommendations from the Standard Technical Specifications for Combustion Engineering Plants, NUREG–1432, Revision 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.

In June 2001, the Nuclear Regulatory Commission (NRC) issued NUREG 1432, Revision 2, ''Standard Technical Specifications Combustion Engineering Plants.'' For Technical Specification 3.7.5, ''Auxiliary Feedwater (AFW) System,'' Surveillance Requirement (SR) 3.7.5.2 requires verification that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head which ensures that AFW pump performance has not degraded during the cycle. This test confirms one point on the pump design curve and is indicative of overall performance. This proposed change will revise San Onofre Nuclear Generating Station (SONGS) Surveillance Frequency to be consistent with NUREG 1432, Revision 2. This change in and of itself will have no effect on the probability or consequences of an accident previously evaluated.

Once this change to the Technical Specification is approved, changes to the Surveillance Frequency of the AFW pumps would be controlled in accordance with the Risk-Informed Inservice Testing Program.

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

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

Response: No.

The proposed amendment will not change the design, configuration or method of operation of the plant. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed amendment will change the SR 3.7.5.2 Frequency from ''31 days on a Staggered Test Basis'' to ''In accordance with the Inservice Testing Program.'' The proposed change does not change the operation or surveillance requirements. It does not change the design function of any of AFW system components. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above, Southern California Edison concludes that the proposed amendment present 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 amendment requests involve no significant hazards consideration.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: December 2, 2002.

Description of amendment request: The proposed amendments change Technical Specification Surveillance Requirement 3.6.4.1.2 to require that only one access door in each access opening of the secondary containment be verified closed every 31 days.

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

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

The proposed change to Surveillance Requirement SR 3.6.4.1.2 would require that only one of the two secondary containment access doors be verified closed; presently, both doors are required to be verified closed. This change is administrative in nature in that it does not involve, require, or result from any physical change to me secondary containment boundary or access door configuration. The change to Surveillance Requirement SR 3.6.4.1.2 is consistent with TSTF Standard Technical Specification Change Traveler TSTF–18, Revision 1, and Surveillance Requirement SR 3.6.4.1.3 of Revision 2 of Volume 1 of NUREG–1433. As indicated in the ''Justification'' portion of Standard Technical Specification Change Traveler TSTF–18, Revision 1, verifying one of the two access doors is closed is sufficient to ensure that the infiltration of outside air does not prevent the establishment and preservation of the required negative pressure within the secondary containment. Indeed, neither the requirements regarding minimum negative pressure and maximum infiltration and drawdown time nor the actions required to be taken should these requirements not be met will be altered by me proposed Licensing amendment.

Because the physical characteristics and performance requirements of the secondary containment will not be altered and the change to Surveillance Requirement SR 3.6.4.1.2 is consistent with the current revision of NUREG–1433, the proposed Licensing amendment can not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

For the reasons previously discussed, neither the secondary containment boundary nor the access door configuration will be altered by or because of the proposed change to the surveillance requirement. Likewise, the requirements defining and governing secondary containment operability and functionality, that is, Standby Gas Treatment system flow rate and secondary containment negative pressure and drawdown limits, will not be changed. The secondary containment, including its access openings, will remain physically unaltered; will function as presently described in the Updated Final Safety Analysis Report [(UFSAR)]; and will be subject to the same structural and functional requirements. Under these circumstances, this change can not, and does not, create the possibility of a new or different kind of accident from any previously evaluated.

3. [Does] the proposed change [* * *] involve a significant decrease in the margin of safety[?]

The requirements defining and governing secondary containment operability and functionality, that is, Standby Gas Treatment system flow rate and secondary containment negative pressure and drawdown limits, will not be changed. The secondary containment, including its access openings will function as presently described in the [* * *] UFSAR and will be subject to the same structural and functional requirements. Therefore, this change can not, and does not, reduce any margin of safety associated with the secondary containment function.

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: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: John A. Nakoski.

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 4, 2002.

Brief description of amendments: The proposed amendments revise several of the Required Actions in the Technical Specifications (TS) that require suspension of operations involving positive reactivity additions or suspension of operations involving reactor coolant system (RCS) boron concentration reductions. In addition, the proposed amendments revise several Limiting Conditions for Operation (LCO) Notes that preclude reductions in RCS boron concentration.

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

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

Response: No.

Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes. The RTS [Reactor Trip System] instrumentation and reactivity control systems will be unaffected. Protection systems will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained.

The probability and consequences of accidents previously evaluated in the FSAR [Final Safety Analysis Report] are not adversely affected because the changes to the Required Actions and LCO Notes assure the limits on SDM [Shutdown Margin] and refueling boron concentration continue to be met, consistent with the analysis assumptions and initial conditions included within the safety analysis and licensing basis. The activities covered by this amendment application are routine operating evolutions. The proposed changes do not reduce the capability of reborating the RCS.

The proposed changes will not involve a significant increase in the probability of any event initiators. The initiating event for an inadvertent boron dilution event, as discussed in FSAR Section 15.4.6, is a failure in the reactor makeup control system (RMCS) or operator error such that inventory makeup with the incorrect boron concentration enters the RCS by way of the CVCS [Chemical and Volume Control System]. Since the RMCS design is unchanged, there will be no initiating event frequency increase associated with equipment failures. However, there could be an increased exposure time per operating cycle to potential operator errors during TS Conditions that, heretofore, prohibited positive reactivity additions. As such, the RTS Instrumentation and RCS Loops TS Bases changes from TSTF [Technical Specification Task Force]-286, Revision 2, have been augmented to preclude the introduction of reactor makeup water into the RCS via the CVCS when one source range neutron flux channel is inoperable or when no RCS loop is in operation. The equipment and processes used to implement RCS boration or dilution evolutions are unchanged and the equipment and processes are commonly used throughout the applicable MODES under consideration. 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.

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

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

Response: No.

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. This amendment will not affect the normal method of plant operation or change any operating limits. The proposed changes merely permit the conduct of normal operating evolutions when additional controls over core reactivity are imposed by the Technical Specifications. The proposed changes do not introduce any new equipment into the plant or alter the manner in which existing equipment will be operated. The changes to operating procedures are minor, with clarifications provided that required limits must continue to be met. No performance requirements or response time limits will be affected. These changes are consistent with assumptions made in the safety analysis and licensing basis regarding limits on SDM and refueling boron concentration.

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

This amendment does not alter the design or performance of the 7300 Process Protection System, Nuclear Instrumentation System, or Solid State Protection System used in the plant protection systems.

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

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

Response: No.

The proposed changes do not alter the limits on SDM or refueling boron concentration. The nominal trip setpoints specified in the Technical Specifications Bases and the safety analysis limits assumed in the transient and accident analyses are unchanged. None of the acceptance criteria for any accident analysis is changed. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor (FDH), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

Therefore, the proposed 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: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

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, 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.*

Carolina Power & Light Company, Docket No. 50–324, Brunswick Steam Electric Plant, Unit 2, Brunswick County, North Carolina

Date of amendment request: September 16, 2002.

Brief description of amendment: The amendment revises a license condition by deleting the requirement to include check valve MVD–V5008 in the facility check valve program.

Date of issuance: December 13, 2002.

Effective date: December 13, 2002.

Amendment No.: 251.

Facility Operating License No. DPR–62: Amendment revises Appendix B, ''Additional Conditions.''

*Date of initial notice in **Federal Register:*** November 12, 2002 (67 FR 68731).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 13, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 31, 2002, as supplemented on September 18, 2002.

Brief description of amendments: The amendments change the method of verifying boron concentration of each safety injection tank. Rather than taking a sample of each tank every 31 days, the revised technical specification surveillance requirement requires leakage into the tanks to be monitored every 12 hours and a sample to be taken every 6 months.

Date of issuance: December 19, 2002.

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

Amendment Nos.: 255 and 232.

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

*Date of initial notice in **Federal Register:*** April 16, 2002. The September 18, 2002, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated December 19, 2002.

No significant hazards consideration comments received: No.

Connecticut Yankee Atomic Power Company, Docket No. 50–213, Haddam Neck Plant, Middlesex County, Connecticut

Date of amendment request: September 10, 2001, as supplemented by letters dated June 19 and November 8, 2002. The supplemental information provided clarification that did not change the scope or the initial no significant hazards consideration determination.

Brief description of amendment: The amendment revises TS 3/4.9.7 and the corresponding Bases to address the use of a single-failure-proof-handling system for the Spent Fuel Building and to remove the restriction on travel of crane loads in excess of 1800 pounds.

Date of issuance: December 17, 2002.

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

Amendment No.: 198.

Facility Operating License No. DPR–61: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 5, 2002 (67 FR 10009).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 17, 2002.

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: August 29, 2002.

Brief description of amendments: The amendments revised the Technical Specifications 3.8.4.7, to modify the note to eliminate the ''once per 60 months'' restriction on replacing the battery service test by the battery modified performance discharge test. Associated changes to the TS Bases are also included.

Date of issuance: December 17, 2002.

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

Amendment Nos.: 209 & 190.

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

*Date of initial notice in **Federal Register:*** November 12, 2002 (67 FR 68733).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 17, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 30, 2002, as supplemented on October 31, 2002.

Brief description of amendment: The amendment revised the requirements in several administrative programs in Technical Specification Section 6.0, ''Administrative Controls.'' Specifically, the amendment: (1) Replaced the specific management titles for several organizational positions with generic titles, (2) replaced the title of the Quality Assurance Program Description with a reference to the quality assurance program described or referenced in the Updated Final Safety Analysis Report, and (3) deleted the functions of the Station Nuclear Safety and the Nuclear Facilities Safety Committees and the Vice President-Nuclear Power since their duties and responsibilities are described in the Quality Assurance Program Description.

Date of issuance: December 17, 2002.

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

Amendment No.: 235.

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

*Date of initial notice in **Federal Register:*** June 25, 2002 (67 FR 42824).

The October 31 supplemental letter provided clarifying information that did not expand the scope of the amendment or change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 17, 2002.

No significant hazards consideration comments received: No.

Indiana Michigan Power Company, Docket No. 50–315, Donald C. Cook Nuclear Plant, Unit 1, Berrien County, Michigan

Date of application for amendment: June 28, 2002, as supplemented on October 15 (two separate letters), October 17, November 15, and December 6, 2002.

Brief description of amendment: The amendment increases the licensed reactor core power level by 1.66 percent from 3250 megawatts thermal (MWt) to 3304 MWt. The power level increase is considered a measurement uncertainty recapture power uprate.

Date of issuance: December 20, 2002.

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

Amendment No.: 273.

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

*Date of initial notice in **Federal Register:*** July 23, 2002 (67 FR 48219).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 20, 2002.

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: October 8, 2002.

Brief description of amendment: The amendment revised Technical Specification 2.7, ''Electrical Systems,'' to increase the amount of diesel fuel oil required for seven days of emergency diesel generator operation.

Date of issuance: December 16, 2002.

Effective date: December 16, 2002, and to be implemented within 30 days of issuance.

Amendment No.: 213.

Facility Operating License No. DPR–40: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** November 12, 2002 (67 FR 68741).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 16, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 24, 2002, as supplemented by letter dated September 24, 2002.

Brief description of amendments: The amendments delete Technical Specification 5.5.3, ''Post Accident Sampling System (PASS),'' and thereby eliminate the requirements to have and maintain the PASS at Plant Hatch.

Date of issuance: December 18, 2002.

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

Amendment Nos.: 235 & 177.

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

*Date of initial notice in **Federal Register:*** August 6, 2002 (67 FR 50958).

The supplement dated September 24, 2002, provided clarifying information that did not change the scope of the June 24, 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 December 18, 2002.

No significant hazards consideration comments received: No

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

Date of amendments request: October 24, 2001, as supplemented by correspondent e-mails dated August 27, 2002, and September 24, 2002.

Brief description of amendments: The amendments consist of relocating various Technical Specifications (TSs) to the Technical Specification Requirements Manual (TRM). The amendments will relocate TSs 3/4.1.3.3, 3/4.3.3.2, 3/4.3.3.11, 3/4.4.7, 3/4.4.9.2, 3/4.3.4.11, 3/4.7.2, 3/4.7.10, 3/4.9.3, 3/4.9.5, 3/4.9.7, 3/4.10.5, and 3/4.11.2.5 to the TRM. Their associated bases will also be relocated to the TRM to be consistent with relocation of the various TSs. In addition, the proposed amendment corrects various typographical and page numbering errors, deletes an outdated one-time exception, and makes minor formal changes to improve consistency.

Date of issuance: The license amendment is effective as of its date of issuance and shall be implemented within 6 months from the date of issuance.

Effective date: December 17, 2002.

Amendment Nos.: Unit 1—145; Unit 2—33.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 5, 2002 (67 FR 5334).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 17, 2002.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket Nos. 50–259, 50–260, and 50–296, Browns Ferry Nuclear Plant, Units 1, 2, and 3, Limestone County, Alabama

Date of application for amendments: September 3, 2002.

Description of amendment request: The proposed amendment revised Surveillance Requirement (SR) 3.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement is added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of issuance: December 23, 2002.

Effective date: Date of issuance, to be implemented within 45 days.

Amendment Nos.: 243, 278, 237.

Facility Operating License Nos. DPR–33, DPR–52, and DPR–68: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** October 15, 2002 (67 FR 63698).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 23, 2002.

No significant hazards consideration comments received: No.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 23, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 3, 2002.

Description of amendment request: The proposed amendment revises Surveillance Requirement (SR) 3.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement is added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of issuance: December 11, 2002.

Effective date: Date of issuance, to be implemented within 45 days.

Amendment No.: 42.

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

Date of initial notice in ***Federal Register:*** October 15, 2002 (67 FR 63699).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 11, 2002.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, et al., Docket No. 50–280, Surry Power Station, Unit 1, Surry County, Virginia

Date of application for amendment: October 15, 2001, as supplemented November 8, 2001, June 28, 2002, and July 25, 2002 .

Brief Description of amendment: This amendment revises the Technical Specifications to allow a one-time change in the Appendix J Type A containment integrated leakage rate test interval from the required 10 years to a test interval of 15 years at Surry Power Station, Unit 1.

Date of issuance: December 16, 2002.

Effective date: December 16, 2002.

Amendment No.: 233.

Facility Operating License No. DPR–32: Amendment changes the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 12, 2001 (66 FR 64309). The November 8, 2001, June 28, 2002, and July 25, 2002, supplements contained clarifying information only and did not change the initial no significant hazards consideration determination or expand the scope of the initial application.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 16, 2002.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 30th day of December 2002.

For the Nuclear Regulatory Commission.

Stuart A. Richards,

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

[FR Doc. 03–156 Filed 1–6–03; 8:45 am]

BILLING CODE 7590–01–P

OVERSEAS PRIVATE INVESTMENT CORPORATION

January 23, 2003 Public Hearing

Time and Date: 1 p.m., Thursday, January 23, 2003.

Place: Offices of the Corporation, Twelfth Floor Board Room, 1100 New York Avenue, NW., Washington, DC.

Status: Hearing open to the public at 1 p.m.

Purpose: Hearing in conjunction with each meeting of OPIC's Board of Directors, to afford an opportunity for any person to present views regarding the activities of the Corporation.

Procedures: Individuals wishing to address the hearing orally must provide advance notice to OPIC's Corporate Secretary no later than 5 p.m. Tuesday, January 21, 2003. The notice must include the individual's name, organization, address, and telephone number, and a concise summary of the subject matter to be presented.

Oral presentations may not exceed ten (10) minutes. The time for individual presentations may be reduced proportionately, if necessary, to afford all participants who have submitted a timely request to participate an opportunity to be heard.

Participants wishing to submit a written statement for the record must submit a copy of such statement to OPIC's Corporate Secretary no later than 5 p.m., Tuesday, January 21, 2003. Such statements must be typewritten, double-spaced, and may not exceed twenty-five (25) pages.

Upon receipt of the required notice, OPIC will prepare an agenda for the hearing identifying speakers, setting forth the subject on which each participant will speak, and the time allotted for each presentation. The agenda will be available at the hearing.

A written summary of the hearing will be compiled, and such summary will be made available, upon written request to OPIC's Corporate Secretary, at the cost of reproduction.

Contact Person for Information: Information on the hearing may be obtained from Connie M. Downs at (202) 336–8438, via facsimile at (202) 218–0136, or via e-mail at *cdown@opic.gov*.

Dated: January 3, 2003.

Connie M. Downs,

OPIC Corporate Secretary.

[FR Doc. 03–313 Filed 1–3–03; 11:17 am]

BILLING CODE 3210–01–M

SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request

[Extension: Rule 17a–7; SEC File No. 270–238; OMB Control No. 3235–0214.]

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

Notice is hereby given that, pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501–3520), the Securities and Exchange Commission (the ''Commission'') has submitted to the Office of Management and Budget (''OMB'') a request for extension of the previously approved collection of information described below.

Rule 17a–7 [17 CFR 270.17a–7] under the Investment Company Act of 1940 (the ''Act'') is entitled ''Exemption of