Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

CONDUCT OF THE MEETING: Dr. Cerqueira, M.D., will chair the meeting. Dr. Cerqueira will conduct the meeting in a manner that will facilitate the orderly conduct of business. The following procedures apply to public participation in the meeting:

(1) Persons who wish to provide a written statement should submit a reproducible copy to Angela McIntosh, U.S. Nuclear Regulatory Commission, Two White Flint North, Mail Stop T8F5, Washington, DC 20555–0001. Hard copy submittals must be postmarked by September 29, 2004. Electronic submittals must be submitted by October 1, 2004. Any submittal must pertain to the topic on the agenda for the meeting.

(2) Questions from members of the public will be permitted during the meeting, at the discretion of the Chairman.

(3) The transcript and written comments will be available for inspection on NRC's Web site *(http://www.nrc.gov)* and at the NRC Public Document Room, 11555 Rockville Pike, Rockville, MD 20852–2738, telephone (800) 397–4209, on or about November 12, 2004. Minutes of the meeting will be available on or about December 17, 2004.

This meeting will be held in accordance with the Atomic Energy Act of 1954, as amended (primarily Section 161a); the Federal Advisory Committee Act (5 U.S.C. App); and the Commission's regulations in Title 10, U.S. Code of Federal Regulations, Part 7.

Dated at Rockville, Maryland, this 21st day of September, 2004.

For the Nuclear Regulatory Commission.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 04–21653 Filed 9–27–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Meetings; Sunshine Act

DATE: Weeks of September 27, October 4, 11, 18, 25, November 1, 2004.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of September 27, 2004

There are no meetings scheduled for the week of September 27, 2004.

Week of October 4, 2004—Tentative

Thursday, October 7, 2004

9:25 a.m. Affirmation Session (Public Meeting) (Tentative).

a. State of Alaska Department of Transportation and Public Facilities (Confirmatory Order Modifying License); appeals of LBP–04–16 by NRC Staff and Licensee (Tentative).

b. Private Fuel Storage (Independent Spent Fuel Storage Installation) Docket No. 72–22–ISFSI (Tentative).

c. USEC, Inc. (Tentative).

10:30 a.m. Discussion of Security Issues (Closed—Ex. 1).

1 p.m. Discussion of Security Issues (Closed—Ex. 1).

2:30 p.m. Discussion of Security Issues (Closed—Ex. 1).

Week of October 11, 2004—Tentative

Wednesday, October 13, 2004

9:30 a.m. Briefing on Decommissioning Activities and Status (Public Meeting) (Contact: Claudia Craig, 301–415–7276).

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

1:30 p.m. Discussion of Intragovernmental Issues (Closed—Ex. 1 & 9).

Week of October 18, 2004—Tentative

There are no meetings scheduled for the week of October 18, 2004.

Week of October 25, 2004—Tentative

There are no meetings scheduled for the week of October 25, 2004.

Week of November 1, 2004—Tentative

There are no meetings scheduled for the week of November 1, 2004.

* 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: Dave 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*

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The NRC provides reasonable accommodation to individuals with disabilities where appropriate. If you need a reasonable accommodation to participate in these public meetings, or need this meeting notice or the transcript or other information from the public meetings in another format (*e.g.* braille, large print), please notify the NRC's Disability Program Coordinator, August Spector, at 301–415–7080, TDD: 301–4152100, or by e-mail at *aks@nrc.gov.* Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

* * * * *

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: September 23, 2004.

Dave Gamberoni,

Office of the Secretary.

[FR Doc. 04–21767 Filed 9–24–04; 9:34 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued, from September 3, 2004, through September 16, 2004. The last biweekly notice was published on September 14, 2004 (69 FR 55466).

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

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

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

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

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

Within 60 days after the date of publication of this notice, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If a request for a hearing or petition for leave to intervene is filed within 60 days, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (First Floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov*.

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

Date of amendments request: July 9, 2004.

Description of amendments request: The amendments would revise the Technical Specifications (TSs) to allow operation of Palo Verde Nuclear Generating Station (PVNGS), Units 1 and 3 up to a maximum reactor core power level of 3990 Megawatts thermal (MWt), an increase of 2.94 percent above the current licensed power level of 3876 MWt. The proposed amendments would also make administrative changes to the PVNGS Unit 2 TSs so that the changed pages would apply to the three PVNGS units. Operation at the uprated power level with replacement steam generators has been approved for PVNGS Unit 2.

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

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

Response: No.

(a) Evaluation of the Probability of Previously Evaluated Accidents

Plant Structures, Systems and Components (SSCs) have been verified to be capable of performing their intended design functions at uprated power conditions. Where necessary, a small number of minor modifications will be made prior to implementation of uprated power operations so that surveillance test acceptance criteria continues to be met. The analysis has concluded that operation at uprated power conditions will not adversely affect the capability or reliability of plant equipment. Current technical specification (TS) surveillance requirements ensure frequent and adequate monitoring of system and component operability. All systems will continue to be operated within current operating requirements at uprated conditions. Therefore, no new structure, system or component interactions have been identified that could lead to an increase in the probability of any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR).

(b) Evaluation of the Consequences of Previously Evaluated Accidents

The radiological consequences were reviewed for all design basis accidents (DBAs) (*i.e.*, both LOCA [loss-of-coolant accident] and non-LOCA accidents) previously analyzed in the UFSAR. The analysis showed that the resultant radiological consequences for both LOCA and non-LOCA accidents remain either unchanged or have not significantly increased due to operation at uprated power conditions. The radiological consequences of all DBAs continue to meet established regulatory limits.

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

Response: No.

The configuration, operation and accident response of the PVNGS [Palo Verde Nuclear Generating Station] Units I and 3 structures, systems, and components are unchanged by operation at uprated power conditions or by the associated proposed TS changes. Analyses of transient events have confirmed that no transient event results in a new sequence of events that could lead to a new accident or different scenario.

The effect of operation at uprated power conditions on plant equipment has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified as a result of operating at uprated conditions. In addition, operation at uprated power conditions does not create any new failure modes that could lead to a different kind of accident. Minor plant modifications, to support Implementation of uprated power conditions, will be made as required to existing SSCs. The basic design function of all SSCs remains unchanged and no new equipment or systems have been installed that could potentially introduce new failure modes or accident sequences.

Based on this analysis, it is concluded that no new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The proposed changes do not have an adverse effect on any safety-related system or design basis function. 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 changes involve a significant reduction in a margin of safety?

Response: No.

A comprehensive analysis was performed to evaluate the effects of power uprate on PVNGS Units 1 and 3. This analysis identified and defined the major input parameters to the NSSS [nuclear steam supply system], reviewed NSSS design transients, and reviewed the capabilities of the NSSS and BOP [balance of plant] fluid systems, NSSS/BOP interfaces, NSSS and BOP control systems, and NSSS and BOP SSCs. All appropriate NSSS accident analyses were re-performed to confirm that acceptable results were maintained and that the radiological consequences remained within regulatory and Standard Review Plan (SRP) limits. The nuclear and thermal hydraulic performance of nuclear fuel was also reviewed to confirm acceptable results. The analyses confirmed that all NSSS and BOP SSCs are capable, some with minor modifications, to safely support operations at uprated power conditions.

The margin of safety of the reactor coolant pressure boundary is maintained under uprated power conditions. The design pressure of the reactor pressure vessel and reactor coolant system will not be challenged as the pressure mitigating systems were confirmed to be sufficiently sized to adequately control pressure under uprated power conditions.

Reanalysis of containment structural integrity under Design Basis Accident (DBA) conditions indicates that the calculated peak containment pressure (Pa) increases from 52.0 psig [pounds per square inch gauge] to 58.0 psig, but remains less than the containment internal design pressure of 60 psig. The proposed value for Pa has been rounded up from the actual calculated value of 57.85 psig.

Radiological consequences of the following accidents were reviewed: Main Steam Line Break, Locked Reactor Coolant Pump (RCP) Rotor, CEA Ejection, Small Steam Line Break Outside Containment, Steam Generator Tube Rupture, LBLOCA, SBLOCA, Waste Gas Decay Tank Rupture, Liquid Waste Tank Failure, and Fuel Handling Accident. The resultant radiological consequences for each of these accidents did not show a significant change due to uprated power conditions and 10 CFR 100 and SRP limits continue to be met.

The analyses supporting operation at power uprate conditions have demonstrated that all systems and components are capable of safely operating at uprated power conditions. All design basis accident acceptance criteria will continue to be met. Therefore, it is concluded that 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 that review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

Attorney for licensee: Kenneth C. Manne, Senior Attorney, Arizona Public Service Company, P.O. Box 52034, Mail Station 7636, Phoenix, Arizona 85072–2034.

NRC Section Chief: Robert Gramm.

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

Date of amendment request: August 19, 2004.

Description of amendment request: The proposed amendment would revise the reactor coolant system (RCS) pressure and temperature limits by replacing Technical Specification Section 3.4.3, ''RCS Pressure and Temperature (P/T) Limits,'' Figures 3.4.3–1 and 3.4.3–2, with figures that are applicable up to 35 effective full-power years (EFPY).

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 RCS P/T limits are based on NRC-approved methodology and will continue to maintain appropriate limits for the HBRSEP [H.B. Robinson Steam Electric Plant], Unit No. 2, RCS up to 35 EFPY. These changes provide appropriate limits for pressure and temperature during heatup and cooldown of the RCS, thus ensuring that the probability of RCS failure is maintained acceptably low. These limits are not directly related to the consequences of accidents.

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 previously evaluated?

The proposed changes will continue to ensure that the RCS will be maintained within appropriate pressure and temperature limits during heatup and cooldown. No physical changes to the HBRSEP, Unit No. 2, systems, structures, or components are being implemented. There are no new or different accident initiators or sequences being created by the proposed Technical Specifications changes. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes ensure that the margin of safety for the fission product barriers protected by these functions will continue to be maintained. This conclusion is based on use of the applicable NRC-approved methodology for developing and establishing the proposed RCS P/T limits. Therefore, these changes do not involve a significant reduction in the margin of safety.

Based on the preceding discussion, the requested change does not involve a significant hazards consideration.

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

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

NRC Section Chief: Michael Marshall (Acting).

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

Date of amendment request: August 11, 2004.

Description of amendment requests: The Haddam Neck Plant (HNP) is currently undergoing active decommissioning. The proposed amendment would revise Technical Specifications (TS) to reflect removal of all Spent Nuclear Fuel (SNF) from the HNP spent fuel pool, and delete the requirement for submittal of an annual Occupational Radiation Exposure Report consistent with Industry's Technical Specifications Task Force (TSTF)–369, Revision 1.

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:

In accordance with 10 CFR 50.92, CYAPCO has reviewed the proposed changes and concluded that the proposed changes do not involve a Significant Hazard Consideration (SHC). The following is provided in support of this conclusion:

Incorporation of TSTF–369, Revision 1: CYAPCO has reviewed the no significant hazards consideration determination published in the **Federal Register** (69 FR 35067) as part of the CLIIP. CYAPCO has concluded that the determination presented in the **Federal Register** is applicable to the HNP and is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91.

Deletion and Relocation of Technical Specifications: The proposed changes do not involve an SHC because the changes would not:

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

The proposed changes (deletion of operational requirements and certain design requirements) reflect the complete transfer of the spent fuel from the spent fuel pool to the Independent Spent Fuel Storage Installation (ISFSI). Design basis accidents related to the spent fuel pool are discussed in the Haddam Neck Plant (HNP) Updated Final Safety Analysis (UFSAR) Chapter 15. These postulated accidents are predicated on spent fuel being stored in the spent fuel pool. With the removal of the spent fuel from the spent fuel pool, there are no remaining safety related Structures, Systems, and Components (SSCs) to be monitored and there are no credible accidents that require the actions of a Certified Fuel Handler or an Equipment Operator to prevent occurrence or mitigate the consequences of an accident.

In addition, the HNP UFSAR Chapter 15 also provides a discussion of other radiological events postulated to occur as a result of decommissioning with the bounding consequences resulting from a fire in a resin container. The proposed changes do not have an adverse impact on decommissioning activities or any of their postulated consequences.

The proposed changes related to the relocation of certain administrative requirements do not affect operating procedures or administrative controls that have the function of preventing or mitigating any design basis accidents. In addition, these proposed changes are consistent with the guidance of NRC Administrative Letter 95–06.

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

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

The proposed changes eliminate the operational requirements and certain design requirements associated with the storage of the spent fuel in the spent fuel pool, and relocate certain administrative controls to the Connecticut Yankee Quality Assurance Program (CYQAP). With the complete removal of the spent fuel from the spent fuel pool, there are no safety related SSCs that remain at the plant. Thus the proposed changes will not have any effect on the operation or design function of safety related SSCs. The proposed changes do not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any previously evaluated.

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

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

NRC Section Chief: Claudia Craig.

Duke Energy Corporation, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: May 27, 2004.

Description of amendment request: The proposed amendment would delete the requirements from the Technical Specifications (TS) to maintain hydrogen recombiners and hydrogen monitors. A notice of availability for the TS improvement using the consolidated line item improvement process was published in the **Federal Register** on September 25, 2003 (68 FR 554416). Licensees were generally required to implement upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide (RG) 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to combustible gas control were imposed by Order for many facilities and were added to or included in the TSs for nuclear power reactors currently licensed to operate.

The revised 10 CFR 50.44, ''Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors,'' eliminated the requirements for hydrogen recombiners and relaxed safety classifications and licensee commitments to certain design and qualification criteria for hydrogen and oxygen monitors.

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated May 27, 2004. *Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration is presented below:

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

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

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

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

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

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

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

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

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

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

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

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

Therefore, this change does not involve a significant reduction in the margin of safety. Removal of hydrogen monitoring from TSs will not result in a significant reduction in their functionality, reliability, and availability.

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

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

NRC Section Chief: Mary Jane Ross-Lee (Acting).

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

Date of amendment request: August 20, 2004.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3.3.8, ''Post Accident Monitoring [PAM] Instrumentation,'' to eliminate TS requirements associated with the reactor building spray (RBS) flow instruments commensurate with the importance of their revised post-accident function.

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

Criterion 1—The proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated

Duke proposes to remove the RBS flow instrument from Technical Specification Table 3.3.8–1 based on a change in its purpose due to recent modifications completed at Oconee. The TS 3.3.8 requirement to declare the affect [affected] RBS System train inoperable is conservative (and inappropriate) when the associated RBS flow instrument is inoperable. Due to recent plant modifications, the RBS flow instruments are no longer needed to allow the operator to throttle flow to preclude RBS pump runout post accident. The revised post accident function of this PAM instrument is to provide information to indicate the operation of the RBS System. There are alternate means to verify that the RBS is in operation, such as, verifying the RBS pump and valve status. The failure of an RBS flow instrument has no impact on the probability of an accident analyzed in the UFSAR [Updated Final Safety Analysis Report]. The RBS flow instrument is no longer needed to mitigate the consequences of an accident analyzed in the UFSAR. As such, the proposed LAR [license amendment request] does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Duke proposes to remove the RBS flow instrument from Technical Specification Table 3.3.8–1 based on a change in its purpose due to recent modifications completed at Oconee. The TS 3.3.8 requirement to declare the affect [affected] RBS System train inoperable is conservative (and inappropriate) when the associated RBS flow instrument is inoperable. Due to recent plant modifications, the RBS flow instruments are no longer needed to allow the operator to throttle flow to preclude RBS pump runout post accident. These changes do not alter the nature of events postulated in the Safety Analysis Report nor do they introduce any unique precursor mechanisms. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3—The proposed amendment would not involve a significant reduction in a margin of safety

The proposed TS changes do not unfavorably affect any plant safety limits, set points, or design parameters. The changes also do not unfavorably affect the fuel, fuel cladding, RCS [reactor coolant system], or containment integrity. Therefore, the proposed TS change, which changes TS requirements associated with revised PAM function of the RBS flow instrument channels, does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Anne W. Cottingham, Winston and Strawn LPP, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Mary Jane Ross-Lee, Acting.

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

Date of amendment request: August 26, 2004.

Description of amendment request: The proposed amendments would add new Technical Specification (TS) 3.3.29 and TS Bases 3.3.29, ''Reactor Building Auxiliary Cooler (RBAC) Isolation Circuitry,'' to accommodate new circuitry that isolates non-safety portions of the low pressure service water (LPSW) system piping inside containment that supply the RBACs. This isolation eliminates potentially damaging water hammers that could occur in the event of certain design-bases events or transients.

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

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

The requested license amendment would add a new Technical Specification to provide appropriate controls for the Reactor Building (RB) Auxiliary Cooler (RBAC) isolation circuitry that is being added to the design of the three Oconee units. The RBAC isolation circuitry provides an automatic means to isolate the LPSW flow stream to the RBACs on a loss of LPSW flow that can lead to a column closure water hammer inside the RB when LPSW flow is restarted. The new circuitry ensures that significant waterhammers do not occur in the LPSW piping to the RBACs and other RB components. The new circuitry will eliminate an Operable but degraded/non-conforming condition associated with potentially damaging waterhammers.

The proposed RBAC isolation circuitry Technical Specification will provide means to assure that the RBAC isolation circuitry operates at a performance level necessary to provide for safe operation of the LPSW system following installation of the LPSW modification and RBAC isolation circuitry at each of the three units. The addition of the RBAC isolation circuitry Technical Specification does not increase the probability or consequences of any accident previously evaluated.

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

The proposed RBAC isolation circuitry Technical Specification provides a means to assure the isolation circuitry operates at a performance level necessary to provide for safe operation of the modified LPSW system flow to the RBACs. The change enhances the plant design by eliminating the possibility of significant waterhammers that could occur inside the RB on a loss of LPSW flow to the RBACs.

The proposed Technical Specification will not create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

Criterion 3—The Proposed Amendment Would Not Involve a Significant Reduction in a Margin of Safety.

The proposed change does not adversely affect any plant safety limits, set points, or design parameters. The change also does not adversely affect the fuel, fuel cladding, Reactor Coolant System, or containment integrity. The RBACs will continue to be isolated during ES events. The modification eliminates significant waterhammers in the LPSW piping to the RBACs.

The change will enhance the ability to provide LPSW flow to safety related loads following LOOP events. Therefore, the proposed change does not involve a reduction in a margin of safety.

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

Attorney for licensee: Anne W. Cottingham, Winston and Strawn LPP, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Mary Jane Ross-Lee, Acting.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: June 24, 2004.

Description of amendment request: The proposed amendment would allow entry into a mode or other specified condition in the applicability of a Technical Specification (TS), while in a condition statement and the associated required actions of the TSs, provided the licensee performs a risk assessment and manages risk consistent with the program in place for complying with the requirements of Title 10 of the Code of Federal Regulations (10 CFR), Part 50, Section 50.65(a)(4). Limiting Condition for Operation (LCO) 3.0.4 exceptions in individual TSs would be eliminated, several notes or specific exceptions would be revised to reflect the related changes to LCO 3.0.4, and Surveillance Requirement (SR) 3.0.4 would be revised to reflect the LCO 3.0.4 allowance.

This change was proposed by the industry's TS Task Force (TSTF) and is designated TSTF–359. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 2, 2002 (67 FR 50475), on possible amendments concerning TSTF–359, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on April 4, 2003 (68 FR 16579). The licensee affirmed the applicability of the following NSHC determination in its application dated June 24, 2004.

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

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

The proposed change allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. Being in a TS condition and the associated required actions is not an initiator of any accident previously evaluated. Therefore, the probability of an accident previously evaluated is not significantly increased. The consequences of an accident while relying on required actions as allowed by proposed LCO 3.0.4, are no different than the consequences of an accident while entering and relying on the required actions while starting in a condition of applicability of the TS. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). Entering into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. The TS allow operation of the plant without the full complement of equipment through the conditions for not meeting the TS LCO. The risk associated with this allowance is managed by the imposition of required actions that must be performed within the prescribed completion times. The net effect of being in a TS condition on the margin of safety is not considered significant. The proposed change does not alter the required actions or completion times of the TS. The proposed change allows TS conditions to be entered, and the associated required actions and completion times to be used in new circumstances. This use is predicated upon the licensee's performance of a risk assessment and the management of plant risk. The change also eliminates current allowances for utilizing required actions and completion times in similar circumstances, without assessing and managing risk. The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard J. Laufer.

Entergy Nuclear Operations, Inc., Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: September 1, 2004.

Description of amendment request: The proposed amendment would delete Technical Specification (TS) 5.6.1, ''Occupational Radiation Exposure Report,'' and TS 5.6.4, ''Monthly Operating Reports.''

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on June 23, 2004 (69 FR 35067). The licensee affirmed the applicability of the model NSHC determination in its application dated September 1, 2004.

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

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

The proposed change eliminates the TS reporting requirements to provide a monthly operating report of shutdown experience and operating statistics if the equivalent data is submitted using an industry electronic database. It also eliminates the Technical Specification reporting requirement for an annual occupational radiation exposure report, which provides information beyond that specified in NRC regulations. The proposed change involves no changes to plant systems or accident analyses. As such, the change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accidents or transients. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This is an administrative change to reporting requirements of plant operating information and occupational radiation exposure data, and has no effect on plant equipment, operating practices or safety analyses assumptions. For these reasons, the proposed change does not involve a significant reduction in the margin of safety.

Based upon the reasoning presented above, the requested change does not involve significance 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, and PSEG Nuclear LLC, Dockets Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3, York and Lancaster Counties, Pennsylvania

Date of application for amendments: June 24, 2004.

Description of amendment request: The proposed amendment would incorporate several Technical Specification Task Force (TSTF) changes to the licensees Technical Specifications (TSs). The specific TSTF changes that would be incorporated are:

(1) TSTF–5, Rev. 1, Delete Safety Limit Violation Notification Requirement—This change modifies TS Section 2.2 to remove the requirements to report safety limit violations. Associated references to Title 10 of the Code of Federal Regulations (10 CFR), Sections 50.72 and 50.73, are also removed.

TSTF–208, Rev. 0, Extension of Time to Reach Mode 2 in LCO (Limiting Condition for Operation) 3.0.3—This TSTF modifies TS Section LCO 3.0.3 to revise the time to be in Mode 2 once LCO 3.0.3 is entered from 7 hours to a bracketed site-specific time depending on the individual plant's ability to reach Mode 2 in a controlled shutdown.

TSTF–222, Rev. 1, Control Rod Scram Time Testing and TSTF–229, Rev. 0, Revise Surveillance Requirement 3.2.2.2 for Consistency with 3.1.4.4—This TSTF modifies the TSs to clarify the frequency of performing control rod scram time testing subsequent to performance of an outage that involved the movement of fuel. The current wording of Surveillance Requirement (SR) 3.1.4.1 could be interpreted that all control rods need to be scram time tested even if the shutdown was for a brief amount of time and only a limited amount of fuel was moved in the reactor (*e.g.,* if only one bundle is moved in a mid-cycle fuel replacement). This change clarifies the intent of the TSs.

TSTF–297, Rev. 1, and TSTF–227, Rev. 0—These two TSTFs affect the following three TS Sections:

3.3.2.2—Feedwater and Main Turbine High Water Level Trip Instrumentation
3.3.4.1—Anticipated Transient Without Scram Recirculation Pump Trip (ATWS–RPT) Instrumentation
3.3.4.2—End of Cycle Recirculation Pump Trip (EOC–RPT) Instrumentation

TSTF–297, Rev. 1—This TSTF modifies the TSs to add a new Required Action and corresponding note to allow affected feedwater pump(s) and main turbine valve(s) to be removed from service. This change is necessary to allow components to be removed from service to fulfill the safety function without a reduction in power to less than 25% rated thermal power. A similar note is added to TS Sections 3.3.4.1 and 3.3.4.2 to provide the same clarification for when the associated Required Action is the appropriate action.

TSTF–227, Rev. 0—This TSTF modifies the TSs to eliminate ambiguity in the EOC–RPT Instrumentation Condition A. Since the LCO allows for having EOC–RPT instrumentation OPERABLE or certain fuel thermal limits are met, Condition A was inappropriately worded. The wording of Condition A is revised to add the word 'required' if one or more channels are inoperable. Without the word 'required', one could interpret Condition A as needing entry even if the fuel thermal limits were being applied instead of applying the operability requirements to the EOC–RPT instrumentation.

TSTF–295, Rev. 0, Post-Accident Monitoring Clarifications—This TSTF modifies the TSs to clarify that a separate Condition entry is allowed for each penetration flow path for the Post Accident Monitoring (PAM) instrumentation Primary Containment Isolation Valve (PCIV) indication function.

TSTF–275, Rev. 0, ECCS Instrumentation Clarifications—This TSTF modifies the TSs to clarify which Emergency Core Cooling System (ECCS) instrumentation is required to be OPERABLE to support Emergency Diesel Generator (EDG) operability. Footnote (a) to Table 3.3.5.1–1 has been changed to only require the affected functions to be OPERABLE in Modes 4 and 5 when the associated ECCS is required to be OPERABLE per LCO 3.5.2.

TSTF–306, Rev. 2, Traversing In-Core Probe Instrumentation Specification Requirements—This TSTF modifies the TSs by adding a note that penetration flow path may not be isolated intermittently under administrative control to conform to what is already allowed for similar specifications for Primary Containment Isolation Valves (PCIVs). Also, the Traversing In-core Probe (TIP) system isolation is set apart as a separate function including the allowance of isolating the penetration instead of requiring a plant shutdown.

TSTF–416, Rev. 0, Clarification of LPCI Operability during Decay Heat Removal Operations—This TSTF modifies the TSs by moving the note that modifies Low Pressure Coolant Injection (LPCI) surveillances to the LCO in LCO 3.5.1 and LCO 3.5.2. These notes provide clarity that the LPCI may be considered OPERABLE during alignment and operation in the decay heat removal Mode.

TSTF–17, Rev. 2, Containment Airlock Testing Frequency—This TSTF modifies the TSs to extend the testing frequency of the containment interlock mechanism from 184 days to 24 months. Also, the corresponding note for this surveillance is no longer required due to the longer surveillance frequency.

TSTF–30, Rev. 3, TSTF–323, Rev. 0, TSTF–45, Rev. 2, TSTF–46, Rev. 1, and TSTF–269, Rev. 2, Containment Isolation Valve Specification Changes—These TSTFs modify TS Sections 3.6.1.3 concerning Primary Containment Isolation Valves (PCIVs) and 3.6.4.2 concerning Secondary Containment Isolation Valves (SCIVs).

TSTF–30, Rev. 3 & TSTF–323, Rev. 0—These TSTFs revise TS 3.6.1.3 to allow for a 72-hour completion time for a closed system flow path with an inoperable isolation valve and allow for a 72-hour completion time for a penetration flow path with an inoperable Excess Flow Check Valve (EFCV).

TSTF–45, Rev. 2—This TSTF revises TSs 3.6.1.3 and 3.6.4.2 to revise surveillance requirements for valve line-ups. Specifically, if a containment isolation valve is locked, sealed, or otherwise secured, they are not required to be verified to be closed during the performance of the surveillance test.

TSTF–46, Rev. 1—This TSTF revises containment isolation valve surveillances to delete the reference to verifying the isolation time of 'each power operated' containment isolation valve and only require verification of each 'automatic isolation valve'.

TSTF–269, Rev. 2—This TSTF allows for verification of valve status by administrative means for repetitive verification of locked, sealed, or secured valves.

TSTF–322, Rev. 2, Secondary Containment Operability Clarification—This TSTF modifies the TSs to clarify the intent of the secondary containment boundary integrity. Associated surveillances currently imply that secondary containment would be inoperable if a Standby Gas Treatment (SGT) subsystem was inoperable.

TSTF–276, Rev. 2, Power Factor for Emergency Diesel Generator (EDG) Surveillances—This TSTF modifies the TSs to allow for certain EDG testing to be performed even if the specified power factor cannot be achieved.

TSTF–65, Rev. 1, Generic Organization Titles—This TSTF modifies the TSs to allow the use of generic organizational titles in place of plant-specific titles. Therefore, for the TSs, a change is requested to replace plant-specific titles with generic titles.

TSTF–299, Rev. 0, Primary Coolant Sources Inspection Requirements—This TSTF modifies the TSs Section 5.2.2, 'Primary Coolant Sources Outside Containment' to clarify the intent of refueling cycle intervals with respect to the system leak test requirements and adds a sentence that the leak test is subject to the provisions of Surveillance Requirements (SR) 3.0.2.

TSTF–279, Rev. 0, Inservice Testing Program Clarifications—This TSTF modifies TSs Section 5.5.8, ''Inservice Testing Program,'' to delete the reference to 'applicable supports' as part of the description for the Inservice Testing Program. The applicable TS Section is 5.5.6.

TSTF–118, Rev. 0, Diesel Generator Fuel Oil Testing Program Clarifications—This TSTF modifies TSs Section 5.5.13, ''Diesel Fuel Oil Testing Program,'' to allow for the provisions of SR 3.0.2 (25% extension) and SR 3.0.3 (missed surveillance actions) to apply to surveillances. The applicable TS Section is 5.5.9.

TSTF–106, Rev. 1, Diesel Generator Fuel Oil Testing Program Clarifications—This TSTF modifies the TSs to clarify that Section 5.5.10.b, concerning verification of the diesel fuel oil that was sampled meets the required ASTM properties, only applies to new fuel. As written, it could be interpreted that this testing is required for existing fuel that is routinely sampled. The applicable TS Section is 5.5.9.b.

TSTF–152, Rev. 0, Routine Reporting Requirements Upgrade—This TSTF modifies the TSs to revise the Occupational Radiation Exposure Report and the Radioactive Effluent Release Report requirements to be consistent with other regulatory changes that have occurred.

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:

A. TSTF–5, Rev. 1, Delete Safety Limit Violation Notification Requirements.

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

Response: No.

This action does not affect the plant or operation of the plant. The change simply removes duplicative information from the Technical Specifications that is covered in the NRC regulations. 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.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system. This change is considered an administrative action to remove duplicative reporting requirements.

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.

This administrative action does not involve any reduction in a margin of safety. Removal of duplicative information does not affect compliance with the regulations. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

B. TSTF–208, Rev. 0, Extension of Time to Reach Mode 2 in LCO 3.0.3.

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

Response: No.

The time frame to take response action in accordance with LCO 3.0.3 is not an initiating condition for any accident previously evaluated and the accident analyses do not assume that any equipment is out of service such that LCO 3.0.3 is entered. The small increase in the time allowed to reach Mode 2 would not place the plant in any significantly increased probability of an accident occurring. The plant would already be proceeding to a plant shutdown condition because of the 1 hour requirement to initiate shutdown actions. There is no change in the time period to reach Mode 3. The Mode 3 Condition is the point where the plant is shutdown. Therefore, since there is no change to the 1 hour requirement to initiate the shutdown nor any change to the time period to reach the shutdown Condition, the small change in the time to reach the Mode 2 status 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.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. There are no plant physical alterations proposed. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system. 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 time period to reach Mode 3 and Mode 4 are unaffected by this activity. This change simply provides a plant specific value for reaching Mode 2 if LCO 3.0.3 is entered which is within the intent of LCO 3.0.3 for performing a controlled plant shutdown. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

C. TSTF–222. Rev. 1, Control, Red Scram Time Testing, and TSTF–229, Rev. 0, Revise Surveillance Requirement 3.2.2.2 for Consistency with 3.1.4.4

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

Response: No.

These changes are considered clarifications to the original intent of the Technical Specifications. Adequate testing of control rods is ensured by this change. Control rod operability is not affected by these changes. 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.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety-related system. 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.

This change is administrative in nature and does not affect any safety analyses assumptions. Adequate control rod testing continues to be maintained with implementation of this activity. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(D) TSTF 297, Rev. 1, and TSTF 227, Rev. 0, Enhancements to Feedwater/Main Turbine High Water Level Trip, EOC–RPT, and ATWS RPT Specifications

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

Response: No.

There are no changes to the plant configuration assumed for any accident. The removal from service of equipment that results in its safety function being met can not adversely affect the consequences of accidents previously evaluated. Other changes are administrative clarifications that have no affect on accidents. 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.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety-related system. 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 actions involved with this activity ensure that safety functions are met. There are no changes in the overall requirements of having trip instrumentation available for event mitigation. There are no affects on the plant safety analyses. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(E) STF–295, Rev. 0, Post-Accident Monitoring Clarications

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

Response: No.

The equipment involved with the revised Technical Specifications are for post-accident monitoring. This equipment has no possibility of increasing the probability of occurrence of the accident since it is monitoring equipment only. The consequences of an accident are not affected since this change maintains the original intent of the Technical Specifications in having available monitoring information for each PCIV penetration path. 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.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system. 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 Technical Specifications continue to require appropriate post accident monitoring equipment to be OPERABLE. Adequate instrumentation for post-accident monitoring will be ensured by the Technical Specification requirements. There are no changes to the plant safety analyses involved with this change. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(F) TSTF–275, Rev. 0, ECCS Instrumentation Clarifications

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

Response: No.

The equipment involved is for mitigative purposes and will not affect the probability of occurrence of an accident. Technical Specifications ensures that adequate mitigative equipment continues to be OPERABLE for any event that may occur in Modes 4 and 5. This change is considered an upgrade to the specifications that will provide more consistency within the Technical Specifications. There are no changes to requirements that ensure appropriate Emergency Core Cooling Systems are OPERABLE. 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.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety-related system.

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.

There is no impact on mitigative equipment that is required to respond to events while in Modes 4 and 5. There is no impact on the plant safety analyses. This change is considered as an upgrade to Technical Specifications that will improve consistency within the Technical Specifications. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(G) TSTF–306, Rev. 2, Traversing In-Core Probe Instrumentation Specifications Requirements

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

Response: No.

The addition of a note that the penetration flow path may be un-isolated under administrative control simply provides consistency with what is already allowed elsewhere in [the] Technical Specifications. The isolation function of the TIP valves are mitigative equipment. They do not create any increased possibility of an accident since they are mitigative. Also, the operation of the manual shear valves is unaffected by this activity. The ability to manually isolate the TIP system by either the normal isolation valve or the shear valve would be unaffected by the inoperable instrumentation. Therefore, the same action as for manual isolation Functions provides an appropriate level of safety. 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.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system.

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 addition of a note that the penetration flow path may be un-isolated under administrative control simply provides consistency with what is already allowed elsewhere in Technical Specifications. The ability to manually isolate the TIP system by either the normal isolation valve or the shear valve would be unaffected by the inoperable instrumentation. Therefore, the same action as for manual isolation Functions provides an appropriate level of safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(H) TSTF–416, Rev. 0 Clarification of LPCI Operability during Decay Heat Removal Operations

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

Response: No.

The proposed change makes the Technical Specifications and their Bases consistent in their consideration of an LPCI subsystem aligned for decay heat removal being considered OPERABLE for ECCS. The LCO 3.5.1 and LCO 3.5.2 Bases state that a LPCI subsystem may be considered OPERABLE during alignment and operation for decay heat removal. As a result, no initiators to accidents previously evaluated are affected and no mitigating equipment assumed in the accidents previously evaluated are affected since the allowance for LPCI being considered operable during these type of shutdown cooling alignments or operations was the intent of the current technical Specifications. Consequently, the probability or consequences of an accident previous evaluated is not significantly increased.

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

Response: No.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system.

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 change makes the Technical Specifications and their Bases consistent in their consideration of an LPCI subsystem aligned for decay heat removal being considered OPERABLE for ECCS. The LCO 3.5.1 and LCO 3.5.2 Bases state that an LPCI subsystem may be considered OPERABLE during alignment and operation for decay heat removal. As the operability requirements of the LPCI subsystem are unaffected, the margin of safety is unaffected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(I) STF–17, Rev. 2, Containment Airlock Testing Frequency

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

Response: No.

The containment airlock is considered as mitigative equipment. Therefore, there are no impacts on the probability of accidents. The proposed surveillance frequency assures that the interlock is working such that there is no unintentional opening of both airlock doors when containment is required. Because the interlock is assured to be working, there will be no significant increase in the consequences of an accident. There is no degradation in the ability of the interlock to assure the containment integrity function is maintained. 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.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system.

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 frequency of 24 months for the interlock testing has been demonstrated to be adequate with regards to the reliability of the airlock. There is no impact on the leak testing requirements. There is no affect on the plant safety analyses. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(J) TSTF–30, Rev. 3, TSTF–323, Rev. 0, TSTF–45, Rev. 2, TSTF–46, Rev. 1, and TSTF–269, Rev. 2, Containment Isolation on Valve Specification Changes

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

Response: No.

The equipment affected by these changes is for mitigative purposes. Therefore, there cannot be an increase in the probability of occurrence of an accident. The controls required in the Technical Specifications are adequate to ensure that the containment barriers are ensured. Isolation valves will be assured to be in their correct positions. Also, inoperable isolation valves in closed systems and inoperable EFCVs have been evaluated to not have any significant impact to the consequences of an accident due to the closed system providing a barrier for the inoperable closed system isolation valve and bounding analyses have been performed for EFCV instrument line failures. 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.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system.

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 equipment affected by these changes is for mitigative purposes. The controls required in the Technical Specifications are adequate to ensure that the containment barriers are ensured. There is no affect on the plant safety analyses. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(K) STF–322, Rev. 2, Secondary Containment Operability Clarification

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

Response: No.

This change involves an administrative clarification to reflect the original intent of the Technical Specifications. There is no impact on the availability of the secondary containment. Additionally, secondary containment is mitigative equipment. 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.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system.

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.

This change involves an administrative clarification to reflect the original intent of the Technical Specifications. There is no impact on the availability of the secondary containment. There is no impact on the plant safety analyses. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(L) TSTF–276, Rev. 2, Power Factor for Emergency Diesel Generator (EDG) Surveillences

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

Response: No.

These changes only affect mitigative equipment and therefore, would not have an impact on the probability of an accident. Also, the performance of the surveillances ensures that mitigative equipment is capable of performing its intended function. 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.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system.

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 performance of the surveillances ensures that mitigative equipment is capable of performing its intended function. There are no degradations in equipment readiness to mitigate design events. There is no adverse affect on the plant safety analysis. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

(M) TSTF–65, Rev. 1, Generic Organizational Titles;

TSTF–299, Rev. 0, Primary Coolant Sources Inspection Requirements;

TSTF–279, Rev. 0, Inservice Testing Program Clarifications;

TSTF–118, Rev. 0, and TSTF–106, Rev. 1, Diesel Generator Fuel Oil Testing Program Clarifications;

TSTF–152, Rev. 0, Routine Reporting Requirement Upgrade

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

Response: No.

The changes to Technical Specification 5.0, Administrative Controls, are considered administrative changes. There are no changes to plant structures, systems or components involved with this change. There are no degradations in the availability of mitigative plant equipment. The proposed changes provide enhancements to the administrative controls in Technical Specifications, therefore, there is no affect on any plant safety analyses; 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.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system.

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 changes to Technical Specification 5.0, Administrative Controls, are considered administrative changes. There are no changes to plant structures, systems or components involved with this change. There are no degradations in the availability of mitigative plant equipment. The proposed changes provide enhancements to the administrative controls in Technical Specifications; therefore, there is no affect on any plant safety analyses. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for Licensee: Thomas S. O'Neill, Associate and General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Section Chief: Daniel S. Collins, Acting.

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

Date of amendment request: August 20, 2004.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) regarding the requirement to demonstrate transfer of the unit A.C. electrical power supply to each offsite circuit and would increase the surveillance exceptions for the A.C. electrical sources in shutdown Modes 5 and 6. Also, the proposed amendment would delete the TS requirement that the auto-connected loads to each emergency diesel generator (EDG) do not exceed the 2000-hour rating of the EDG.

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

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

No. The proposed surveillance requirement changes do not alter the design or operation of any structure, system, or component. No previously analyzed accident scenario is changed. Initiating conditions and assumptions remain as previously analyzed. The revised surveillance requirements will continue to assure adequate performance of structures, systems, and components. 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?

No. The proposed surveillance requirement changes do not alter the design or operation of any structure, system, or component. No new or different accident initiators are created as a result of the proposed changes. Therefore, the proposed changes do 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?

No. The proposed surveillance requirement changes do not reduce or adversely affect the capabilities of the offsite and onsite electrical power sources. The revised surveillance requirements will continue to assure adequate performance of structures, systems, and components. The proposed changes do not affect conformance of the electrical power systems to the applicable design criteria. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The Nuclear Regulatory Commission (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.

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

Date of amendment request: September 1, 2004.

Description of amendment request: The proposed amendments would revise the Operating Licenses' licensing basis to allow use of the code for Generation of Thermal-Hydraulic Information for Containment, Version 7.1patch1 (GOTHIC 7) to model Prairie Island Nuclear Generating Plant (PINGP) containment response for loss of coolant accidents (LOCA) and main steam line break (MSLB) accidents. The current PINGP containment response analyses are performed utilizing CONTEMPT. The Nuclear Management Company is making this request to support a transition option from internal analyses using CONTEMPT to an external analyses vendor (Westinghouse), which supports GOTHIC 7.

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

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

Response: No.

The proposed amendment will change the Prairie Island Nuclear Generating Plant licensing basis by allowing use of the Generation of Thermal-Hydraulic Information for Containment, Version 7.1patch1, to model containment response for loss of coolant accident (LOCA) and main steam line break (MSLB) accidents.

The containment is not an accident initiator, thus changing the containment modeling methodology does not increase the probability of an accident. This license amendment proposes to use a new methodology for modeling containment response analyses following an accident inside containment involving release of steam and water. This amendment does not alter the nuclear reactor core or reactor coolant system equipment, nor does it alter the methods or equipment used directly in mitigation of an accident. Thus radioactive releases inside containment due to an accident and radioactive releases from containment are not affected by the proposed change in analysis methodology. As discussed in Exhibits C and D, the Gothic 7 sample results for the LOCA and MSLB transients predicted that the containment would remain below design pressure for both cases. Therefore, this change does not increase the consequences of an accident previously evaluated.

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

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

Response: No.

The proposed amendment will change the Prairie Island Nuclear Generating Plant licensing basis by allowing use of the Generation of Thermal-Hydraulic Information for Containment, Version 7.1patch1, to model containment response for LOCA and MSLB accidents.

The proposed amendment does not involve changes to plant design, hardware, system operation, or procedures involved with containment function. The proposed changes include application of new methodology for analysis of containment response following a loss of coolant accident or steam line break accident. The results of the analyses are used to demonstrate that the acceptance criteria for the containment structure continue to be met. These changes do not create the possibility for a new or different kind of accident.

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 amendment will change the Prairie Island Nuclear Generating Plant (PINGP) licensing basis by allowing use of the Generation of Thermal-Hydraulic Information for Containment, Version 7.1patch1 (GOTHIC 7), to model containment response for LOCA and MSLB accidents.

The proposed licensing basis change to use GOTHIC 7 affects the design basis LOCA and MSLB containment accident analyses. As discussed in Exhibits C and D, the GOTHIC 7 sample results for the LOCA and MSLB transients predicted that the containment would remain below design pressure for both cases. The GOTHIC 7 accuracy in this application has been verified through benchmark analyses against the current analyses of record, validated against recognized standard data, and found to be appropriate for application to the PINGP design basis accidents. Safety analysis acceptance criteria are satisfied and adherence to safety analysis acceptance criteria using GOTHIC 7 assures that Technical Specification limits will not be exceeded during normal operation. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: L. Raghavan.

South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of amendment request: May 21, 2004.

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

The NRC staff issued a notice of availability of a model no significant hazards consideration determination for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated May 21, 2004.

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

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

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

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

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

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

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

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

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

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

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

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

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

Therefore, this change does not involve a significant reduction in the margin of safety. Removal of hydrogen monitoring from TS will not result in a significant reduction in their functionality, reliability, and availability.

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

Attorney for licensee: Thomas G. Eppink, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief (Acting): Mary Jane Ross-Lee.

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 amendment requests: August 26, 2004.

Description of amendment requests: The proposed amendments would revise the Technical Specifications (TS) to implement ZIRLO™ fuel rod cladding material into the fuel design for San Onofre Nuclear Generating Station (SONGS), Units 2 and 3. Specifically, the licensee requests to add reference to ZIRLO™ clad fuel and filler rods in TS 4.2.1, ‘‘Fuel Assemblies,’’ and in TS 5.7.1.5, ‘‘Core Operating Limits Report (COLR),’’ add the following references to the list of analytical methods used to determine the core operating limits: ‘‘Calculative Methods for the C–E Nuclear Power Large Break LOCA [loss-of-coolant accident] Evaluation Model,’’ CENPD–1 32, Supplement 4–P–A, August 2000, and ‘‘Implementation of ZIRLO™ Cladding Material in CE [Combustion Engineering, Inc.] Nuclear Power Fuel Assembly Designs,’’ CENPD–404–P–A, November 2001.

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

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

Response: No.

The proposed change allows the use of methods required for the implementation of ZIRLO™ clad fuel rods in San Onofre Nuclear Generating Station (SONGS) Units 2 and 3. The use of this methodology will not increase the probability of an accident because the plant systems will not be operated outside of design limits, no different equipment will be operated, and system interfaces will not change.

As ZIRLO™ material is introduced to the reactor, transition cores will exist in which fuel assemblies containing ZIRLO™ and Zircaloy clad fuel rods are co-resident. Each type of fuel assembly (ZIRLO™ or Zircaloy clad fuel rods) will be evaluated based on the approved topical reports listed in TS 5.7.1.5.

The use of this additional methodology will not increase the consequences of an accident because Limiting Conditions of Operation (LCOs) will continue to restrict operation to within the regions that provide acceptable results, and Reactor Protection System (RPS) trip setpoints will restrict plant transients so that the consequences of accidents will be acceptable. In addition, the consequences of the accidents will be calculated using NRC accepted methodologies.

The transition cores that will exist as ZIRLO™ clad fuel is introduced to the reactor will not increase the consequences of an accident. Operation within the LCOs and RPS setpoints will continue to restrict plant transients so that the consequences of accidents will be acceptable.

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 change does not add any new equipment, modify any interfaces with any existing equipment, alter the equipment's function, or change the method of operating the equipment. The proposed change does not alter plant conditions in a manner that could affect other plant components. The proposed change does not cause any existing equipment to become an accident initiator. The ZIRLO™ clad fuel rod design does not introduce features that could initiate an accident.

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

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

Response: No.

Safety Limits ensure that Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded during steady state operation, normal operational transients and anticipated operational occurrences. All fuel limits and design criteria shall be met based on the approved methodologies defined in the topical reports. The RPS in combination with the LCOs will continue to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a violation of the Safety Limits. Therefore, the proposed changes will have no impact on the margins as defined in the Technical Specification bases.

The safety analyses determine the LCO settings and RPS setpoints that establish the initial conditions and trip setpoints, which ensure that the Design Basis Events (Postulated Accidents and Anticipated Operational Occurrences) analyzed in the Updated Final Safety Analysis Report (UFSAR) produce acceptable results. In addition, all fuel limits and design criteria shall be satisfied. The Design Basis Events that are impacted by the implementation of ZIRLO™ cladding will be analyzed using the NRC accepted methodology described in CENPD–404–P–A.

The change in the fuel rod cladding material and the use of the Emergency Core Cooling System (ECCS) performance evaluation models, CENPD–132, Supplement 4–P–A, ''Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model'' and CENPD–137, Supplement 2–P–A, ''Calculative Methods for the ABB [Asea Brown Boveri] CE Small Break LOCA Evaluation Model'' will not involve a reduction in the margin of safety because LCOs and Limiting Safety System Settings (LSSS) will be adjusted, if necessary, to maintain acceptable results for the impacted Design Basis Events.

Therefore, this 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: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Robert Gramm.

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: May 21, 2004.

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

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated May 21, 2004.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

NRC Section Chief: Mary Jane Ross-Lee, Acting.

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: July 20, 2004.

Description of amendment request: The proposed amendments would revise Administrative Controls Section 5.3.1 to replace the specific designation for the Health Physics Superintendent with a reference to the senior individual in charge of Health Physics, and to add flexibility to the qualification requirements for unit staff positions. This change supports Southern Nuclear Company's ongoing initiative to achieve fleet standardization.

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 Technical Specifications Administrative Controls Section 5.3.1 involves the use of a more generic designation for the unit staff position responsible for Health Physics without reducing the level of authority required for that position. The proposed change also allows the flexibility to use an NRC accredited program for qualifying personnel to fill unit staff positions, which represents an acceptable alternative to the qualification requirements for these positions as currently specified in the Technical Specifications. Since the proposed changes are administrative in nature, they do not involve any physical changes to any structures, systems, or components, nor will their performance requirements be altered. The proposed changes also do not affect the operation, maintenance, or testing of the plant. Therefore, the response of the plant to previously analyzed accidents will not be affected. Consequently, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes to the Technical Specifications will have no adverse impact on the overall qualification of the unit staff. The alternative use of an accredited program that has been endorsed by the NRC will ensure the educational requirements and power plant experience for each unit staff position are properly satisfied and will continue to fulfill applicable regulatory requirements. Also, since no change is being made to the design, operation, maintenance, or testing of the plant, no new methods of operation or failure modes are introduced by the proposed changes. Therefore, the possibility of a new or different kind of accident from any previously evaluated is not created.

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

The proposed changes to the Technical Specifications will have no adverse impact on the onsite organizational features necessary to assure safe operation of the plant. Lines of authority for plant operation are unaffected by the proposed changes. Also, the adoption of the more generic designation of the individual responsible for Health Physics will reduce the regulatory burden of having to devote limited resources to process a license amendment whenever a title change for this position is implemented. Accordingly, this reduction in regulatory burden and the option to use an accredited program endorsed by NRC to qualify the unit staff will improve plant efficiency without compromising plant safety. Therefore, the proposed changes do not involve a significant decrease 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: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Mary Jane Ross-Lee, Acting.

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

Date of amendment request: May 21, 2004.

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

The NRC staff issued a notice of availability of a model no significant hazards consideration determination (NSHC) for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated May 21, 2004.

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

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

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

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

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

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

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

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

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

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

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

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

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

Therefore, this change does not involve a significant reduction in the margin of safety. Removal of hydrogen monitoring from TS will not result in a significant reduction in their functionality, reliability, and availability.

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

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Section Chief: Mary Jane Ross-Lee, Acting.

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

Date of amendment request: May 21, 2004.

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

The NRC staff issued a notice of availability of a model no significant hazards consideration determination (NSHC) for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated May 21, 2004.

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

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

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

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

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

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

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

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

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

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

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

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

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

Therefore, this change does not involve a significant reduction in the margin of safety. Removal of hydrogen monitoring from TSs will not result in a significant reduction in their functionality, reliability, and availability.

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

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Section Chief: Mary Jane Ross-Lee, Acting.

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

Date of amendment request: August 26, 2004.

Description of amendment request: The license amendment request proposes revising the Technical Specifications (TSs) to delete the TS requirements related to Hydrogen Analyzers and Hydrogen Recombiners consistent with NRC-approved TS Task Force (TSTF) Traveler number TSTF–447, Revision 1, ''Elimination of Hydrogen Recombiners and Change to Hydrogen and Oxygen Monitors.'' The TS requirements related to Hydrogen Analyzers and Hydrogen Recombiners are contained in TS Tables 3.3–10 and 4.3–10 and TSs 3.6.4.1 and 3.6.4.2. The availability of this TS improvement was announced in the **Federal Register** on September 25, 2003, as part of the Consolidated Line Item Improvement Process (CLIIP).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The analysis endorses the NRC staff's generic no significant hazards consideration determination for TSTF–447 which was published in the **Federal Register** on September 25, 2003 (68 FR 55416) as follows:

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

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

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

The regulatory requirements for the hydrogen monitors can be relaxed without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. Classification of the hydrogen monitors as Category 3, and removal of the hydrogen monitors from TS will not prevent an accident management strategy through the use of the SAMGs [Severe Accident Management Guidelines], 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 the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, does not involve a significant increase in the probability or the consequences of any accident previously evaluated.

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

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

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

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

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

The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. Category 3 hydrogen monitors are adequate to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The intent of the requirements established as a result of the TMI, Unit 2 accident can be adequately met without reliance on safety-related hydrogen monitors.

Therefore, this change does not involve a significant reduction in the margin of safety. Removal of hydrogen monitoring from TS will not result in a significant reduction in their functionality, reliability, and availability.

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

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

NRC 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, Public File Area 01F21, 11555 Rockville Pike (First Floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov*.

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 amendment: August 6, 2002, as supplemented December 12, 2002, July 24, 2003, and March 1, May 20, and August 11, 2004.

Brief description of amendment: The amendments replace the Technical Specifications 3.9.4 and 3.9.5 requirements to close all containment penetrations providing direct access from the containment atmosphere to outside temperature with a set of more detailed and less restrictive requirements.

Date of issuance: September 13, 2004.

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

Amendment No.: 268 and 244.

Renewed Facility Operating License No. DPR–53: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** October 15, 2002 (67 FR 63690).

The December 12, 2002, July 24, 2003, March 1, 2004, and May 20, 2004, letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The August 11, 2004, letter withdrew the licensee's requested changes to Technical Specification 3.9.3.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 13, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 12, 2003.

Brief description of amendments: The amendments delete Technical Specification Section 5.5.3, ''Post-Accident Sampling.''

Date of issuance: September 15, 2004.

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

Amendment Nos.: 269 and 245.

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

Date of initial notice in ***Federal Register:*** April 13, 2004 (69 FR 19564).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 7, 2002, as supplemented April 7, 2003 and July 19, 2004.

Brief description of amendment: The amendment relocates the boration system Technical Specification (TS) requirements to the Technical Requirements Manual and the boron dilution analysis restrictions within the TSs. The amendment also revises the TS limiting condition for operation action and the surveillance requirements associated with the emergency core cooling, containment spray and cooling and auxiliary feedwater systems.

Date of issuance: September 9, 2004.

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

Amendment No.: 283.

Facility Operating License No. DRP–65: The amendment revised the TSs.

Date of initial notice in ***Federal Register:*** June 11, 2002 (67 FR 40021). The April 7, 2003, and July 19, 2004, supplements contained clarifying information and did not change the staff's initial proposed finding of no significant hazards consideration.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 7, 2002, as supplemented November 5, 2003.

Brief description of amendment: The amendment revises the Technical Specifications (TSs) related to safety system settings. Specifically, the amendment revises: (1) TS 1.0 ''Definitions;'' (2) TS 2.2.1 ''Limiting Safety System Settings—Reactor Trip System Instrumentation Setpoints;'' (3) TS 3.3.1 ''Reactor Trip System Instrumentation;'' (4) TS 3.3.2 ''Engineered Safety Features Actuation System Instrumentation;'' (5) TS 3.7.7 ''Control Room Emergency Ventilation System;'' and (6) TS 3.8.3.1 ''Onsite Power Distribution—Operating.''

Date of issuance: September 14, 2004.

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

Amendment No.: 220.

Facility Operating License No. DRP–49: The amendment revised the TSs.

Date of initial notice in ***Federal Register:*** October 15, 2002 (67 FR 63692). The November 5, 2003, supplement contained clarifying information and did not change the staff's initial proposed finding of no significant hazards consideration.

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

No significant hazards consideration comments received: No.

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

Date of application of amendments: May 25, 2004.

Brief description of amendments: The amendments revised the licensing basis in the Updated Final Safety Analysis Report (UFSAR) to support installation of a low-pressure injection (LPI) cross connect inside containment. The changes to the UFSAR revise the licensing basis for selected portions of the core flood and LPI/Decay Heat Removal piping to allow exclusion of the dynamic effects associated with postulated rupture of that piping by application of leak-before-break technology. The amendments also revise the Technical Specifications (TSs) to delete TSs that will no longer apply when the LPI cross connect modification has been implemented.

Date of issuance: September 2, 2004.

Effective date: As of the date of issuance and shall be implemented during the fall 2004 refueling outage of Unit 3.

Amendment Nos.: 340, 342, and 341.

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

*Date of initial notice in **Federal Register:*** July 6, 2004 (69 FR 40673). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 2, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 20, 2004, as supplemented by letters dated May 19, July 13, and August 16, 2004.

Brief description of amendments: The amendments change the Prairie Island technical specification (TS) on containment to implement a portion of TSs Task Force Traveler 5, ''Revise containment requirements during handling irradiated fuel and core alterations.'' The amendments also selectively implement an alternative source term per Title 10 of the Code of Federal Regulations, Section 50.67 to perform the radiological consequences analysis of the design-basis fuel handling accident which supports the proposed TS changes.

Date of issuance: September 10, 2004.

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

Amendment Nos.: 166 and 156.

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

*Date of initial notice in **Federal Register:*** May 25, 2004 (69 FR 29769).

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

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

No significant hazards consideration comments received: No.

Pacific Gas and Electric Company, Docket No. 50–133, Humboldt Bay Power Plant, Unit 3, Humboldt County, California

Date of application for amendment: June 23, 2004.

Brief description of amendment: The amendment removes a restriction from the Humboldt Bay Power Plant Unit 3 license thereby permitting Pacific Gas and Electric to engage in active decommissioning of the facility.

Date of issuance: September 10, 2004.

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

Amendment No.: 35.

Facility Operating License No. DPR–7: This amendment revises the license.

*Date of initial notice in **Federal Register:*** August 3, 2004 (69 FR 46587).

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

No significant hazards consideration comments received: No.

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

Date of amendments request: June 28, 2004, as supplemented by letter dated August 5, 2004.

Brief Description of amendments: The amendments revise TS 3.4.13, ''RCS [Reactor Coolant System] Operational Leakage,'' TS 5.5.9, ''Steam Generator [SG] Tube Surveillance Program,'' and TS 5.6.10, ''Steam Generator Tube Inspector Report.'' They also add a new TS 3.4.17, ''Steam Generator Tube Integrity.'' These changes facilitate implementation of industry initiative NEI [Nuclear Energy Institute] 97–08, ''Steam Generator Program Guidelines,'' which allows a comprehensive, performance-based approach to managing SG performance at Farley Nuclear Plant, Units 1 and 2.

Date of issuance: September 10, 2004.

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

Amendment Nos.: 163 and 156.

Facility Operating License Nos. NPF–2 and NPF–8: Amendments revise the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 3, 2004 (69 FR 46950). The supplemental letter dated August 5, 2004, provided clarifying information that did not change the initial proposed no significant hazards consideration determinations.

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 6, 2002, as supplemented by letters dated December 19, 2002, March 28, June 24, September 3, and October 22, 2003.

Brief description of amendment: The amendment revised the Technical Specifications (TSs) to (1) relocate the pressure temperature limit curves and low temperature overpressure protection system limits to the Pressure and Temperature Limits Report (PTLR), (2) reference the PTLR in the affected TSs limiting conditions for operation and bases, including the addition of the PTLR to the definitions section of the TSs, and the addition of a new TS 6.9.1.15 to the administrative controls section of the TSs, (3) relocate TS 3.4.9.2, Pressurizer, to the Sequoyah Technical Requirements Manual and (4) revise TS 3.4.9.1, Pressure/Temperature Limits, Reactor Coolant System, and TS 3.4.12, Low Temperature Over Pressure Protection Systems, to incorporate standard TSs requirements from NUREG–1431, Revision 2, ''Standard Technical Specifications—Westinghouse Plants.''

Date of issuance: September 15, 2004.

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

Amendment Nos.: 294 and 284.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 29, 2002 (67 FR 66015). The supplemental letters provided clarifying information that did not expand the scope of the original application 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 September 15, 2004.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 17th day of September, 2004.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

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BILLING CODE 7590–01–P

PRESIDENT'S COUNCIL ON INTEGRITY AND EFFICIENCY

Senior Executive Service Performance Review Board Membership

AGENCY: Environmental Protection Agency (EPA).

ACTION: Notice.

SUMMARY: This notice sets forth the names and titles of the current membership of the PCIE Performance Review Board as of September 23, 2004.

EFFECTIVE DATE: September 28, 2004.

FOR FURTHER INFORMATION CONTACT: Individual Offices of (the) Inspector General.

SUPPLEMENTARY INFORMATION:

I. Background

The Inspector General's Act of 1978, as amended, has created independent audit and investigative units—Offices of (the) Inspector General—at 57 Federal agencies. In 1981, the President's Council on Integrity and Efficiency (PCIE) was established by Executive Order as an interagency committee charged with promoting integrity and effectiveness in Federal programs. The PCIE is chaired by the Office of management and Budget's Deputy Director for Management, and comprised principally of the 29 Presidential appointed Inspectors General (IGs). The primary objectives of the PCIE are: (1) Mounting collaborative efforts to address integrity, economy, and effectiveness issues that transcend individual Federal agencies; and (2) increasing the professionalism and effectiveness of IG personnel throughout the Government.

II. PCIE Performance Review Board

Under 5 U.S.C. 4314(c)(1)–(5), and in accordance with regulations prescribed by the Office of Management and Budget, each agency is required to establish one or more Senior Executive Service (SES) performance review boards. The purpose of these boards is to review and evaluate the initial appraisal of a senior executive's performance by the supervisor, along with any recommendations to the appointing authority relative to the performance of the senior executive. The current members of the President's Council on Integrity and Efficiency Performance Review Board, as of September 23, 2004, were as follows:

Agency for International Development

Phone Number: (202) 712–1170; PCIE/ECIE Liaison—Donna Rosa (202) 712–4993

James R. Ebbitt—Deputy Inspector General
Adrienne Rish—Assistant Inspector General for Investigation
Robert S. Perkins—Counsel to the Inspector General
Bruce Crandlemire—Assistant Inspection General for Audit
Paula Hayes—Assistant Inspector General for Management

DEPARTMENT OF AGRICULTURE

Phone Number: (202) 720–8001 PCIE/ECIE Liaison—Cheryl Viani (202) 720–8001

Joyce N. Fleischman—Deputy Inspector General
Tracy A. LaPoint—Deputy Assistant Inspector General
David R. Gray—Counsel to the Inspector General
Suzanne M. Murrin—Assistant Inspector General for Policy Development and Resources Management
Mark R. Woods—Assistant Inspector General for Investigations
Jon E. Novak—Deputy Assistant Inspector General for Investigations
Robert W. Young, Jr.—Assistant Inspector General for Audit
Marlane T. Evans—Deputy Assistant Inspector General for Audit

DEPARTMENT OF COMMERCE

Phone Number: (202) 482–4661 PCIE/ECIE Liaison—Allison Lerner (202) 482–1577

Edward L. Blansitt—Deputy Inspector General
Anthony D. Mayo—Assistant Inspector General for Investigation
Elizabeth T. Barlow—Counsel to the Inspector General
Judith J. Gordon—Assistant Inspector General for Systems Evaluation
Jill A. Gross—Assistant Inspector General for Inspections and Program Evaluation
Jessica Rickenbach—Assistant Inspector General for Compliance and Administration

DEPARTMENT OF DEFENSE

Phone Number: (703) 604–8324 PCIE/ECIE Liaison—John R. Crane (703) 604–8324

Charles W. Beardall—Director, Defense Criminal Investigative Service—Office of the Deputy Inspector General for Investigations
Patricia Brannin—Assistant Inspector General for Audit Policy and Oversight, Office of the Deputy Inspector General for Inspections and Evaluations
John R. Crane—Assistant Inspector General for Communications and Congressional Liaison
Thomas Gimble—Deputy Inspector General for Intelligence
Donald Horstman—Director, Investigations of Senior Officials, Office of the Deputy Inspector General for Investigations
Francis E. Reardon—Deputy Inspector General for Auditing
Mary Ugone—Assistant Inspector General, Acquisition Management, Office of the Deputy Inspector General for Auditing
Keith West—Assistant Inspector General, Audit Followup and Technical Support, Office of the Deputy Inspector General for Auditing
Daniel F. Willkens—Deputy Director, Defense Criminal Investigative Service, Office of the Deputy Inspector General for Investigations
Shelton R. Young—Assistant Inspector General, Readiness and Logistics Support, Office of the Deputy Inspector General for Auditing

DEPARTMENT OF EDUCATION

Phone Number: (202) 205–6900 PCIE/ECIE Liaison—Kira Stankosky (202) 245–6997

Thomas Carter—Deputy Inspector General
Cathy Lewis—Assistant Inspector General for Evaluations, Inspections and Management Services
Helen Lew—Assistant Inspector General for Audit Services
George Rippey—Deputy Assistant Inspector General for Audit Services
Thomas Sipes—Assistant Inspector General for Investigative Services
Charles Coe—Assistant Inspector General for Information Technology and Computer Crimes Investigation
Mary Mitchelson—Counsel to the Inspector General

DEPARTMENT OF ENERGY

Phone Number: (202) 586–4393 PCIE/ECIE Liaison—Arlene Acton (202) 586–1807

John Hartman—Assistant Inspector General for Investigations
Rickey Hass—Assistant Inspector General for Audit Operations
Denise Smith—Assistant Inspector General for Resource Management
Christopher Sharpley—Deputy Inspector General for Investigations and Inspections
Linda Snider Director for Audit Policy and Administration Sanford Parnes Counsel to the Inspector General

DEPARTMENT OF HEALTH AND HUMAN SERVICES

Phone Number: (202) 619–3148 PCIE/ECIE Liaison—Sheri Denkensohn (202) 619–3148

Lewis Morris—Chief Counsel to the Inspector General
Tony Campbell—Assistant Inspector General for Operations Division, Office of Investigations
Donald Dille—Acting Deputy Inspector General for Management and Policy
Joe Green—Assistant Inspector General for Audit Management and Policy