(Public Meeting) (Contact: John Larkins, 301–415–7360)

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

Week of July 26, 2004—Tentative

There are no meetings scheduled for the Week of July 26, 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.

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SUPPLEMENTARY INFORMATION:

By a Vote of 3–0 on June 9, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Affirmation of 1) Private Fuel Storage (Independent Spent Fuel Storage Installation) Docket No. 72–22–ISFSI'' be held on June 9, and no less than one week's notice to the public.

By a vote of 3–0 on June 15, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Affirmation of 1) Request to Export up to 140 Kilograms of Weapons-Grade Plutonium Oxide (PuO_2) to Cogema's Cardarache and Melox Facilities in France (XSNM03327)'' be held on June 15, and on less than one week's notice to the public.

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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 case-by-case basis.

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

Dated: June 17, 2004.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 04–14160 Filed 6–18–04; 9:47 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, May 28, 2004, through June 10, 2004. The last biweekly notice was published on June 8, 2004 (69 FR 32070).

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

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

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

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

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed by: (1) First class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff; (2) courier, express mail, and expedited delivery services: Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff; (3) e-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, *hearingdocket@nrc.gov;* or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC, Attention: Rulemakings and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

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

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

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: May 5, 2004.

Description of amendment request: The proposed change will revise Technical Specification Surveillance Requirement (SR) 4.0.5.a for inservice inspection (ISI) and testing of American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components, to include a reference to the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) in addition to Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by Title 10 of the Code of Federal Regulations (10 CFR), Section 50.55a(g).

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

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

The proposed changes to the Technical Specification SR 4.0.5.a and the associated Bases are requested to add a reference to the ASME OM Code and applicable Addenda for inservice inspection of ASME Code Class 1, 2, and 3 components.

The existing Technical [Specification] requires inservice inspection of ASME Code Class 1, 2, and 3, components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves as required by 10 CFR 50.55a. The purposes of the inservice inspection and inservice testing programs are to assess the operational readiness of pumps and valves, to detect degradation that might affect component operability, and to maintain safety margins with provisions for increased surveillance and corrective action. 10 CFR 50.55a defines the requirements for applying industry codes and standards to each licensed nuclear power facility. The initial HNP [Shearon Harris Nuclear Power Plant, Unit 1] ISI program was developed in accordance with NRC regulations (10 CFR 50.55a(g)(4)(i)) to comply with the 1983 Edition of the ASME Boiler and Pressure Vessel Code, including Addenda through the Summer of 1983 and is reflected in the existing Technical Specifications and associated Bases sections.

The current, second ten-year interval HNP ISI program was developed in accordance with the 1989 Edition (no Addenda) of ASME Boiler and Pressure Vessel Code, Section XI. Subarticles IWF–1200 and IWF–5300 require the examination and testing of snubbers per the first Addenda of ASME/ANSI [American National Standards Institute] OM–1987, Part 4 (published in 1988), generally referred to as ''OM–4.'' HNP Relief Request 2RG–008, Revision 1, grants HNP the ability to retain the snubber testing and examination program in Technical Specification 3/4.7.8.

The 1995 Edition with 1996 Addenda of the ASME OM Code, Subsection ISTD, is the applicable Code per Code Case OMN–13. HNP plans to utilize the 1995 Edition with 1996 Addenda of the ASME OM Code for snubber visual examinations as an approved alternative to the snubber visual examination requirements of the 1989 Edition of ASME Section XI and as modified by HNP Relief Request 2RG–008, Revision 1. Code Case OMN–13 has been evaluated and approved by the NRC in Reg Guide 1.192.

The proposed change to Technical Specification SR 4.0.5.a is also administrative in nature. The proposed changes comply with approved codes and standards. As a result, there will be no affect on plant safety.

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

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

The changes to Technical Specification SR 4.0.5.a and Bases section 4.0.5 and are being proposed to reference the ASME OM Code in addition to Section XI of the ASME Boiler and Pressure Vessel Code. The proposed changes are administrative in nature and do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility.

The use of the ASME OM Code 1995 Edition with 1996 Addenda, Subsection ISTD, with incorporation of the snubber visual examination frequency of Code Case OMN–13 will result in an improvement in personnel safety and dose reduction.

This change will have no operational impact, therefore, the proposed change 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 changes to Technical Specification SR 4.0.5.a and Bases section 4.0.5 do not involve a reduction in the margin of safety. As previously identified, the subject changes are administrative in nature and will add a reference to the ASME OM Code in Technical Specification SR 4.0.5.a. Therefore, the proposed changes to the Technical Specifications and Bases will not result in a reduction in the margin of safety.

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

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

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

Date of amendment request: January 30, 2004.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.3.6.2, ''Secondary Containment Isolation Instrumentation,'' Condition C, to add the words, ''not met,'' to the end of the sentence, ''Required Action and associated Completion Time.'' The omission of the words, ''not met,'' was an oversight during the change to the Improved Standard Technical Specifications (ISTS), NUREG 1433.

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

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

Response: No.

The proposed change corrects the sentence in Condition C of TS 3.3.6.2 by indicating that when this condition is not met, certain actions are required. This terminology is prevalent throughout the ISTS and is implied in this section as well. No changes in operating practices or physical plant equipment are created as a result of this terminology addition. 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 type of accident from any accident previously evaluated?

Response: No.

This proposed change is a correction of an action statement in TS 3.3.6.2. No physical change in plant equipment will result from this proposed change. Therefore, the proposed change does not create the possibility of a new or different type 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 is editorial in nature and only provides a correction to an action statement in the Secondary Containment Isolation Instrumentation involving inoperable channels and automatic functions to agree with NUREG 1433. 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: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: March 19, 2004.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.3.6.1, ''Primary Containment Isolation Instrumentation,'' to correct a formatting error introduced during conversion to Improved Technical Specifications (ITS) by replacing ''1 per room'' with ''2'' for the Required Channels Per Trip System for the Reactor Water Cleanup (RWCU) Area Ventilation Differential Temperature—High primary containment isolation instrumentation.

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

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

Response: No.

The proposed change restores the number of Required Channels Per Trip System of the RWCU Area Ventilation Differential Temperature—High isolation, Function 5.c of Table 3.3.6.1–1 of TS 3.3.6.1, Primary Containment Isolation Instrumentation, to its pre-ITS value and adds an explanatory note. No changes in operating practices or physical plant equipment are created as a result of this change. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change restores the number of Required Channels Per Trip System of the RWCU Area Ventilation Differential Temperature—High isolation, Function 5.c of Table 3.3.6.1–1 of TS 3.3.6.1, Primary Containment Isolation Instrumentation, to its pre-ITS value and adds an explanatory note. No physical change in plant equipment will result from this proposed change. Therefore, the proposed change does not create the possibility of a new or different type 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 is administrative in nature and only provides a correction to Table 3.3.6.1–1 of TS 3.3.6.1, Primary Containment Isolation Instrumentation, as well as an explanatory note. 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: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Section Chief: L. Raghavan.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, Washington

Date of amendment request: May 19, 2004.

Description of amendment request: The proposed change revises Technical Specification (TS) 3.8.1, ''AC Sources—Operating,'' to permit a longer completion time for the Division 1 and Division 2 diesel generators (DGs). This is a risk-informed TS change that would extend the DG completion time from 72 hours (the current limit) to 14 days.

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

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

The proposed change does not adversely affect the design of the DGs, the operational characteristics or function of the DGs, the interfaces between the DGs and other plant systems, or the reliability of the DGs. Required Actions and the associated Completion Times are not initiating conditions for any accident previously evaluated, and the DGs are not initiators of any previously evaluated accidents.

The DGs support the mitigation of the consequences of previously evaluated accidents that involve a loss of offsite power. The consequences of a previously analyzed accident will not be significantly affected by the extended DG Completion Time since the remaining DGs will continue to be capable of performing their accident mitigation function as assumed in the accident analysis. Thus, the consequences of accidents previously analyzed are unchanged between the existing TS requirements and the proposed changes. The consequences of an accident are independent of the time the DGs are out of service as long as there are adequate DGs available.

Based on the above, the proposed change to extend the DG allowed Completion Time during plant operation will not involve a significant increase in accident probabilities or consequences.

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

No new accidents would be created since no changes are being made to the plant that would introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators; neither does it adversely impact any accident mitigating systems. The addition of an independent AACSBC [alternate AC source to the Division 1 and Division 2 battery chargers] will provide added time for responding to a loss of all AC power assumed in the accident analyses. The design of the AACSBC will contain features and administrative controls to maintain the separation and protection of emergency AC distribution systems and does not create the possibility of a new or different kind of accident from any previously evaluated.

Based on the above, implementation of the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident. These barriers include the fuel cladding, the reactor coolant system, and the containment system. Throughout the period of the current TS Completion Time, when one DG is out-of-service during power operation, the margin of safety is managed by limiting the allowed outage time and other concurrent power source outages within the TS. This time period is a temporary relaxation of the single failure criteria, which, consistent with overall system reliability considerations, provides a limited time to repair the equipment and conduct testing. The extension of the current TS Completion Time to 14 days has been determined not to be a significant reduction in the margin of safety. The proposed changes will not result in a significant decrease in DG availability so that the assumptions regarding DG availability are not impacted. Probabilistic Risk Assessment (PRA) methods, and a deterministic analysis were utilized to fully evaluate the effect of the proposed DG Completion Time extension. The results of the analysis show no significant increase in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF). Energy Northwest has proposed a number of risk management actions to reduce the possibility of a plant transient; a loss of high-pressure injection and cooling systems, a loss of other on-site power sources, or a loss of offsite power during the period the DG is out-of-service.

Based on the above, the change to the TS Completion Time does not result in a significant reduction in the margin of safety. This is based on our management of plant risk, the reliability of the other diesel generators, and the inclusion of risk management actions.

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 C. Poindexter, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: May 12, 2004.

Description of amendment request: The proposed amendment would change the reactor core analytical methods used to determine the core operating limits, reflect the changes allowed by Technical Specification Task Force (TSTF) Traveler No. 363, ''Revised Topical Report References in ITS [Improved Standard Technical Specifications] 5.6.5, COLR [Core Operating Limits Report],'' and delete the Index from the Technical Specifications (TSs).

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

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

Response: No.

TS 6.9.5.1, Core Operating Limits Report (COLR)

The proposed amendment, in part, identifies a change in the nuclear physics codes used to confirm the values of selected cycle-specific reactor physics parameter limits and includes minor editorial changes which do not alter the intent of stated requirements. The proposed change also allows the use of methods required for the implementation of ZIRLO clad fuel rods. Inasmuch as the proposed change includes codes that have been previously approved by the NRC [Nuclear Regulatory Commission] for CE [Combustion Engineering] cores, the amendment is administrative in nature and has no impact on any plant configuration or system performance relied upon to mitigate the consequences of an accident. Parameter limits specified in the COLR for this amendment are not changed from the values presently required by TSs. Future changes to the calculated values of such limits may only be made using NRC approved methodologies, must be consistent with all applicable safety analysis limits, and are controlled by the 10 CFR 50.59 process. Assumptions used for accident initiators and/or safety analysis acceptance criteria are not altered by this change.

The proposed change also implements NRC approved TSTF Traveler No. 363. This is an administrative change that will allow specific details, such as the revision number, revision date, and supplement number of topical reports that are referenced in the TSs, to be deleted and relocated in the cycle specific COLR. This proposed change does not result in any changes to the assumptions used to evaluated accident initiators and/or safety analysis acceptance criteria.

Index

The proposed deletion of the Index is purely administrative and does not impact the accident analysis.

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

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

Response: No.

TS 6.9.5.1, Core Operating Limits Report (COLR)

The proposed change, in part, identifies a change in the nuclear physics codes used to confirm the values of selected cycle-specific reactor physics parameter limits. The proposed change also allows the use of methods required for the implementation of ZIRLO clad fuel rods. Neither of these changes results in a change to the physical plant or to the modes of operation defined in the facility license.

The proposed change also implements TSTF Traveler No. 363. The proposed change does not result in changes to the physical plant or to the modes of operation defined in the facility license nor does it involve the addition of new equipment or the modification of existing equipment.

Index

The proposed deletion of the Index is purely administrative has no affect on existing equipment.

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

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

Response: No.

TS 6.9.5.1, Core Operating Limits Report (COLR)

The proposed changes to change the nuclear physics code package and to add a topical report to support the use of ZIRLO do not amend the cycle specific parameter limits located in the COLR from the values presently required by the TS. The individual specifications continue to require operation of the plant within the bounds of the limits specified in COLR. Benchmarking has shown that uncertainties for the Westinghouse Physics code system yields are essentially the same or less than those obtained for the current ROCS/DIT methodology. Future changes to the values of these limits by the licensee may only be developed using NRC approved methodologies, must remain consistent with all applicable plant safety analysis limits addressed in the Safety Analysis Report, and are further controlled by the 10 CFR 50.59 process. The relocation of the supplement numbers, revision numbers, and approval dates of the analytical methods listed in the COLR does not affect the margin of safety. The analysis will continue to be performed using NRC approved methodology. Safety analysis acceptance criteria are not being altered by this amendment.

Index

The proposed deletion of the Index, which is an administrative document, does not impact any TS values or safety limits.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: February 27, 2004.

Description of amendment request: This amendment request incorporates a revision to the Technical Specifications and licensing and design bases that supports a full-scope application of an Alternative Source Term (AST) methodology.

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 Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below:

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

Adoption of the AST and those plant systems affected by implementation of the AST do not initiate design-basis accidents (DBAs). The proposed changes do not affect the design or manner in which the facility is operated; rather, once the occurrence of an accident has been postulated, the new AST is an input to analyses that evaluate the radiological consequences. Therefore, the proposed changes do not involve an increase in the probability of an accident previously evaluated.

The structures, systems and components (SSCs) affected by the proposed change act as mitigators to the consequences of accidents. Based on the revised analyses, the proposed changes do revise certain performance requirements; however, the proposed changes involve different acceptance criteria. There cannot, therefore, be a direct comparison to determine if the proposed change would result in an increase in consequences over the current design. However, the licensee's analysis proposes that, with implementation of AST, all regulatory acceptance criteria continue to be met. Therefore, any potential increase in consequences would not be considered significant.

Therefore, the proposed changes do not result in 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?

Implementation of AST does not affect the design function or mode of operations of SSCs in the facility prior to a postulated accident. Since SSCs are operated essentially the same after the AST implementation, no new failure modes are created by this proposed change.

Therefore, the proposed changes will 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 a margin of safety?

The changes proposed are associated with a revision to the licensing basis. These changes would modify the input to DBA analyses from the original source term to the AST. Based on the revised analyses, the proposed changes involve different acceptance criteria. There cannot, therefore, be a direct comparison to determine if the proposed change would result in a reduction in a margin of safety. However, the licensee's analysis proposes that, with implementation of AST, all regulatory acceptance criteria continue to be met. The dose consequences of the accident analyses revised in support of the proposed changes are subject to the acceptance criteria in 10 CFR 50.67, ''Accident source term,'' Regulatory Guide 1.183, ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,'' and Standard Review Plan 15.0.1, ''Radiological Consequence Analyses Using Alternative Source Terms.'' Thus, by meeting the applicable regulatory limits for AST, any potential decrease in a margin of safety would not be considered significant.

Therefore, because the proposed changes continue to result in dose consequences within the applicable regulatory limits, the changes are considered to not result in a significant reduction in a margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: April 8, 2004.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS), Section 6, Administrative Controls, to relocate (1) the Plant Operations Review Committee and Nuclear Review Board requirements, (2) the program/procedure review and approval requirements, and (3) the record retention requirements to the Quality Assurance Topical Report, the document controlling the licensee's quality assurance program.

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

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

Response: No. The proposed changes involve the relocation of several administrative requirements from the Technical Specifications (TS) to a document subject to the control of 10 CFR 50.54(a), and is therefore, administrative in nature. The relocated requirements involve the onsite and offsite organization's review and audit, the review and approval of procedures, and the retention of records. The change will not alter the physical design or operational procedures associated with any plant structure, system, or component. The change does not reduce the duties and responsibilities of the organizations performing the review, audit, and approval functions essential to ensuring the safe operation of the plant.

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

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

Response: No. The proposed changes are administrative in nature. The changes do not alter the physical design, safety limits, or safety analysis assumptions, associated with the operation of the plant. Accordingly, the changes do not introduce any new accident initiators, nor do they reduce or adversely affect the capabilities of any plant structure, system, or component to perform their safety function.

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

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

Response: No. The proposed changes conform to NRC regulatory guidance regarding the content of plant Technical Specifications. The guidance is presented in Administrative Letter 95–06, and NUREG–1433, Rev. 2. The relocation of these administrative requirements will not reduce the quality assurance commitments as accepted by the NRC, nor reduce administrative controls essential to the safe operation of the plant. Future changes to these administrative requirements will be performed in accordance with NRC regulation 10 CFR 50.54(a), consistent with the guidance identified above. Accordingly, the relocation results in an equivalent level of regulatory control.

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

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

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

NRC Section Chief: James W. Clifford.

FPL Energy Seabrook, LLC, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: March 17, 2004.

Description of amendment request: The proposed amendment would revise the operating license and Technical Specifications (TSs) to support an increase in the licensed power from 3411 megawatts thermal (MWt) to 3587 MWt. This represents an increase of approximately 5.2 percent above the current rated licensed thermal power.

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 Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below:

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

Plant structures, systems and components (SSCs) have been verified to be capable of performing their intended design functions at uprated power conditions. Where necessary, some components will be modified prior to implementation of uprated power operations to accommodate the revised operating conditions. The analysis indicated that operation at uprated power conditions will not adversely affect the capability of plant equipment. Current TS surveillance requirements ensure frequent and adequate monitoring of system and component operability. All systems will continue to be operated in accordance with current design requirements under uprated conditions; therefore, no new components or system 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).

The radiological consequences were reviewed for design basis accidents (DBAs) previously analyzed in the UFSAR. The analysis showed that the resultant radiological consequences for both loss-of-coolant accidents (LOCAs) and non-LOCAs remain either unchanged or have increased due to operation at uprated power conditions. Any increase in the radiological consequences of DBAs is not considered significant because plant operation at uprated power conditions continue to meet established regulatory limits.

Therefore, the proposed changes do not result in 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?

The configuration, operation, and accident response of the SSCs 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 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 systems and components. The basic design function of all SSCs remains unchanged and no new safety-related equipment or systems will be installed which 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 TS changes do not have an adverse effect on any safety. Therefore, the proposed changes will 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 a margin of safety?

A comprehensive analysis was performed to support the power uprate program at the Seabrook Station. This analysis identified and defined the major input parameters to the Nuclear Steam Supply System (NSSS), reviewed NSSS design transients, and reviewed the capabilities of the NSSS fluid systems, NSSS/BOP (balance-of-plant) interfaces, and NSSS and BOP components. The nuclear and thermal hydraulic performance of nuclear fuel was also reviewed to confirm acceptable results. Only minor plant modifications, to support implementation of uprated power conditions, will be made as required to existing systems and components. Changes in setpoints for actuation of equipment do not adversely affect the outcome of any postulated accident. The analysis indicated that all NSSS and BOP systems and components will continue to operate within existing design and safety limits 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.

The radiological consequences were reviewed for DBAs previously analyzed in the UFSAR. The analysis showed that the radiological consequences of DBAs continue to meet established regulatory limits at uprated power conditions.

The analyses supporting the power uprate program have demonstrated that all systems and components are capable of safely operating at uprated power conditions. All DBA acceptance criteria will continue to be met. Therefore, it is concluded that the proposed changes do not result in a significant reduction in the margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: May 27, 2004.

Description of amendment request: The proposed amendment would revise the Cooper Nuclear Station (CNS) Technical Specifications (TS). The proposed amendment would lower the reactor vessel water level at which the reactor water cleanup (RWCU) system isolates, secondary containment isolates, and the control room emergency filter system (CREFS) starts. General Electric (GE) Service Information Letter (SIL) No. 131 discussed problems that result from isolation of the RWCU and start of the standby gas treatment (SGT) system, in conjunction with isolation of secondary containment. The SIL recommended that the vessel water level at which these actions occur be lowered, thereby eliminating these problems and the resulting unnecessary complications with scram recovery. The proposed changes to the CNS TS are in accordance with SIL 131 Recommendations 1 and 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?

No. The values of various plant parameters at which piping connected to the reactor vessel and containment isolates and air-filtering systems start are not accident precursors. Thus, lowering the reactor vessel water level at which RWCU and secondary containment isolate and SGT and CREFS initiate has no impact on the probability of a design basis accident evaluated in the CNS Station Safety Analysis. Therefore the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed logic changes involve no changes to the logic of the Reactor Protection System that initiates automatic reactor shutdown in response to an accident. The proposed logic changes involve no changes to the logic of the Emergency Core Cooling System (ECCS) that initiates automatic actions to ensure adequate core cooling and containment integrity in response to an accident. The CNS response to the design basis accidents (DBAs) addressed in the Station Safety Analysis with the proposed changes to the logic was evaluated. This evaluation has demonstrated that there is no increase in the offsite radiological doses to the public resulting from these accidents.

Based on the above NPPD [Nebraska Public Power District] concludes that 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?

The proposed lowering of the level of water in the reactor vessel at which certain automatic actions would occur changes the operation of various systems at CNS. However, the change in system operation is not significant. Currently automatic actions occur in the RWCU System, SGT System, CREFS, and secondary containment in response to reactor vessel water level. Changing the level at which these automatic actions occur is not a significant change in the systems operation. Hardware changes needed to implement the modified logic are minor. Lowering the reactor vessel water level for these actions does not introduce a new mode of plant operation and does not create a potential for any new failure mechanisms, malfunctions, or accident initiators. Making this change does not involve adding new systems to the CNS design.

Based on the above NPPD concludes that 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?

The safety margin associated with dose consequences to the public following DBAs is based on, in part, automatic operation of systems that shut down the reactor, automatic initiation of ECCS, and automatic isolation of primary and secondary containment. The proposed changes to the CNS TS make no changes that affect the automatic shutdown of the reactor or the automatic initiation and operation of ECCS. The plant response to DBAs with the proposed revisions to the RWCU isolation (primary containment) and the SGT and the CREFS initiation (secondary containment) have been evaluated and shown to not result in any increase in dose to the public. The safety margin associated with dose consequences to the control room operators is based on automatic isolation of secondary containment, and initiation of CREFS. The plant response to DBAs with the proposed revisions to the RWCU isolation (primary containment) and SGT and CREFS initiation (secondary containment) have been evaluated and shown to not result in any increase in dose to the control room operators.

Based on the above NPPD concludes that the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: May 14, 2004.

Description of amendment request: The proposed amendment will relocate the requirements of Technical Specification (TS) 3.3(1)a, ''Reactor Coolant System and Other Components Subject to ASME XI Boiler & Pressure Vessel Code Inspection and Testing Surveillance,'' concerning inservice inspection of ASME Class 1, 2, and 3 components and TS 3.4, ''Reactor Coolant System Integrity Testing,'' concerning reactor coolant system integrity testing to the Fort Calhoun Station (FCS) Updated Safety Analysis Report (USAR). These TSs do not meet the criterion in 10 CFR 50.36(c)(2)(ii) for inclusion in the TSs.

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

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

The proposed amendment relocates the requirements of TS 3.3(1)a concerning inservice inspection of ASME Class 1, 2, and 3 components and TS 3.4 concerning reactor coolant system integrity testing to the FCS USAR. These TSs are directed toward prevention of component degradation and continued long term maintenance of acceptable structural conditions. It is not necessary to retain these TSs to ensure immediate operability of safety systems. Therefore these TSs do not meet the criteria set forth in 10 CFR 50.36(c)(2)(ii) for inclusion in the TS. The requirements are being relocated from TS to the FCS USAR, which will be maintained pursuant to 10 CFR 50.59, thereby reducing the level of regulatory control. [This reduction in the] level of regulatory control has no impact on the probability or consequences of an accident previously evaluated. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change relocates requirements of TS 3.3(1)a concerning inservice inspection of ASME Class 1, 2, and 3 components and TS 3.4 concerning reactor coolant system integrity testing that do not meet the criteria for inclusion in TS set forth in 10 CFR 50.36(c)(2)(ii). The change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change will not impose different requirements, and adequate control of information will be maintained. This change will not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change relocates requirements of TS 3.3(1)a concerning inservice inspection of ASME Class 1, 2, and 3 components and TS 3.4 concerning reactor coolant system integrity testing that do not meet the criteria for inclusion in TS set forth in 10 CFR 50.36(c)(2)(ii). The change will not reduce a margin of safety since the location of a requirement has no impact on any safety analysis assumptions. In addition, the relocated requirements of TS 3.3(1)a and TS 3.4 concerning inservice inspection and testing of ASME Class 1, 2, and 3 components remain the same as the existing TS. Since any future changes to these requirements will be evaluated per the requirements of 10 CFR 50.59, there will be no reduction in a margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: May 21, 2004.

Description of amendment request: The proposed amendment would add information to the Technical Specification (TS) Basis for TS 2.4, ''Containment Cooling,'' to allow containment spray pumps to be secured during a loss-of-coolant accident (LOCA) to minimize the potential for containment sump clogging when certain conditions are met. NRC Bulletin 2003–01, ''Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors,'' required that operators of pressurized water reactor (PWR) plants state that the emergency core cooling systems (ECCS) and the containment spray (CS) recirculation functions meet applicable regulatory requirements with respect to adverse post-accident debris blockage or describe interim compensatory measures to reduce the risk associated with the potentially degraded or non-conforming ECCS and CS recirculation functions.

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

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

The proposed changes will not [significantly] increase the probability or consequences of any accident based on the following:

The proposed compensatory action is only taken following a LOCA if all safeguards have functioned and if an excess of CS flow exists above that required to control containment pressure, temperature, and remove the accident source term. The proposed action is only taken if the worst-case single failure has not occurred indicating maximum containment cooling and SI [safety injection] flow delivered, and minimum source term due to no severe core damage. The proposed action occurs following the peak containment pressure transient, therefore, the action has no impact on the peak containment pressure analysis. A quantitative analysis of the change in LOCA consequences due to suspension of CS flow for 10 minutes has not been performed. However, the prerequisite conditions for taking this action provide reasonable assurance that the loss of the remaining CS train for ten minutes will not result in a significant increase in the LOCA consequences. Therefore, the proposed changes will not [significantly] increase the probability or consequence of any accident.

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

The proposed revision does not involve physical changes to any equipment required to mitigate the consequences of an accident, nor alter how design basis accident events are postulated. The proposed change alters the method of controlling an Engineered Safety Feature following a design basis event so that manual actions are substituted for automatic actions. Reasonable assurance exists that these manual actions can be taken in a timely manner to allow continued CS system operation to provide containment cooling and source term reduction with no significant increases in the radiological consequences or approaching of design containment limits. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change alters the method of controlling an Engineered Safety Feature following a design basis event so that manual actions are substituted for automatic actions. The proposed actions are only taken following a LOCA if all safeguards have functioned and if an excess of CS flow exists above that required to control containment pressure, temperature, and remove the accident source term. The prerequisite conditions for taking this action provide reasonable assurance that the loss of the remaining CS train will not result in a reduction in the margin of safety for radiological consequences or containment design parameters. Therefore, the proposed changes do not involve a significant reduction to 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: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment requests: March 18, 2004.

Description of amendment requests: The proposed amendments would authorize updates of the Diablo Canyon Power Plant (DCPP) Final Safety Analysis Report (FSAR) Update to use on a permanent basis, a revised steam generator (SG) voltage-based repair criteria probability of detection (POD) method using plant specific SG tube inspection results, referred to as the probability of prior cycle detection (POPCD) method.

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

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

The use of a revised steam generator (SG) voltage-based repair criteria probability of detection (POD) method, the probability of prior cycle detection (POPCD) method, to determine the beginning of cycle (BOC) indication voltage distribution for the Diablo Canyon Power Plant (DCPP) Units 1 and 2 operational assessments does not increase the probability of an accident. Based on industry and plant specific bobbin detection data for outside diameter stress corrosion cracks (ODSCC) within the SG tube support plate (TSP) region, large voltage bobbin indications which individually can challenge structural or leakage integrity can be detected with near 100 percent certainty. Since large voltage ODSCC bobbin indications within the SG TSP can be detected, they will not be left in service, and therefore these indications should not be included in the voltage distribution for the purpose of operational assessments. The POPCD method improves the estimate of potentially undetected indications for operational assessments, but does not directly affect the inspection results. Since large voltage indications are detected, they will not result in an increase in the probability of a steam generator tube rupture (SGTR) accident or an increase in the consequences of a SGTR or main steam line break (MSLB) accident.

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

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

The use of the POPCD method to determine the BOC voltage distribution for the DCPP Units 1 and 2 operational assessments concerns the SG tubes and can only affect numerical predictions of probabilities for the SGTR accident. Since the SGTR accident is already considered in the Final Safety Analysis Report Update, there [is] no possibility to create a design basis accident that has not been previously evaluated.

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

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

The use of the POPCD method to determine the BOC voltage distribution for the DCPP Units 1 and 2 operational assessments does not involve a significant reduction in a margin of safety. The applicable margin of safety potentially impacted is the Technical Specification 5.6.10, ''Steam Generator (SG) Tube Inspection Report,'' projected end-of-cycle leakage for a MSLB [main steam line break] accident and the projected end-of-cycle probability of burst. Based on industry and plant specific bobbin detection data for ODSCC within the SG TSP region, large voltage bobbin indications that can individually challenge structural or leakage integrity can be detected with near 100 percent certainty and will not be left in service. Therefore these indications should not be included in the voltage distribution for the purpose of operational assessments. Since these large voltage indications are detected, they will not result in a significant increase in the actual end-of-cycle leakage for a MSLB accident or the actual end-of-cycle probability of burst. The POPCD method approach to POD considers the potential for missing indications that might challenge structural or leakage integrity by applying the POPCD data from successive inspections. If a large indication was missed in one inspection, it would continue to grow until finally detected in a later inspection.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: April 27, 2004.

Description of amendment request: The proposed change will revise the Safety Limit Minimum Critical Power Ratio (SLMCPR) values for two recirculation loop and one recirculation loop operation. Each safety limit value will be applicable for all fuel types in the Hope Creek Generating Station core. In the amendment request, PSEG Nuclear LLC requested changes to the Technical Specifications to support the use of GE14 fuel and General Electric Company (GE) reload analysis methods beginning with the upcoming Cycle 13.

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 SLMCPR ensures that no mechanistic fuel damage occurs in the core if the limit is not violated. The revised SLMCPR values maintain the appropriate conservative margin to boiling transition and the probability of fuel damage is not increased. The derivation of the revised SLMCPR values specified in the Technical Specifications has been performed using NRC approved methods and uncertainties. The analysis methodology incorporates appropriate cycle-specific parameters and uncertainties in determining the revised SLMCPR values. The analyses do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient. The revised SLMCPR values do not affect the performance of systems or components used to mitigate the consequences of accidents previously evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or radiological 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 revised SLMCPR values specified in the Technical Specifications have been calculated in accordance with NRC approved methods and uncertainties. The changes do not involve any new method for operating the facility and do not involve any facility modifications. No new initiating events or anticipated operational occurrences result from these changes.

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

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

Response: No.

The revised SLMCPR values are calculated using NRC approved methods and uncertainties. The revised SLMCPR values continue to ensure that greater than 99.9% of all fuel rods in the core are expected to avoid boiling transition if the safety limits are not violated, thereby maintaining the fuel cladding integrity during normal plant operation and anticipated operational occurrences.

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: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Clifford.

PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: April 26, 2004.

Description of amendment request: The proposed change will revise the Salem Unit Nos. 1 and 2 source term used for design basis radiological analysis, in accordance with the provisions of 10 CFR 50.67, ''Accident Source Term''. The proposed change will also revise certain requirements in the Technical Specifications (TSs) and the Updated Final Safety Analysis Report (UFSAR) based on the radiological dose analysis margins obtained in the Alternate Source Term application.

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

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

The alternative source term analysis does not change the design of the plant or affect the performance of the systems or components used to mitigate the consequences of accidents previously evaluated. The analyses do not change the method of operating the plant and has no effect on the probability of an accident initiating event or a transient. The alternative source term calculations demonstrate the radiological consequences to the design basis accidents specified in the plant's UFSAR will still remain well below the radiological limits specified in 10 CFR 100.11. Therefore, since the radiological consequences are well below the specified limits and the probability of an accident is unchanged, 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?

The proposed amendment is not the result of a hardware design change, nor does it lead to the need for a hardware design change. There is no change in the methods or procedures by which the unit is operated. As a result, all structures, systems, and components will continue to perform as previously analyzed by the licensee, and previously evaluated and accepted by the NRC staff. Therefore, the proposed amendment will 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?

The proposed changes result in operation in accordance with regulatory guidelines and support the revisions to the radiological analysis of the limiting design basis accidents. The radiological consequences of these accidents are all within the regulatory acceptance criteria associated with the use of the alternative source term methodology. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the NRC staff's analysis, 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: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Clifford.

Rochester Gas and Electric Corporation, Docket No. 50–244, R.E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: March 1, 2004.

Description of amendment request: The proposed amendment would extend the completion time (CT) from 1 hour to 24 hours for Condition B of Technical Specification (TS) 3.5.1, ''Accumulators.'' The accumulators are part of the emergency core cooling system and consist of tanks partially filled with borated water and pressurized with nitrogen gas. The contents of the tank are discharged to the reactor coolant system (RCS) if, as during a loss-of-coolant accident, the coolant pressure decreases to below the accumulator pressure. Condition B of TS 3.5.1 specifies a CT to restore an accumulator to operable status when it has been declared inoperable for a reason other than the boron concentration of the water in the accumulator not being within the required range. This change was proposed by the Westinghouse Owners Group participants in the TS Task Force (TSTF) and is designated TSTF–370. TSTF–370 is supported by NRC-approved Topical Report WCAP–15049-A, ''Risk-Informed Evaluation of an Extension to Accumulator Completion Times,'' submitted on May 18, 1999. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on July 15, 2002 (67 FR 46542), on possible amendments concerning TSTF–370, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 12, 2003 (68 FR 11880). The licensee affirmed the applicability of the following NSHC determination in its application dated March 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 basis for the accumulator limiting condition for operation (LCO), as discussed in Bases Section 3.5.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of the WCAP–15049, ''Risk-Informed Evaluation of an Extension to Accumulator Completion Times,'' evaluation, the proposed change will allow plant operation with an inoperable accumulator for up to 24 hours, instead of 1 hour, before being required to begin shutdown. The impact of the increase in the accumulator CT on core damage frequency for all the cases evaluated in WCAP–15049 is within the acceptance limit of 1.0E–06/yr for a total plant core damage frequency (CDF) less than 1.0E–03/yr. The incremental conditional core damage probabilities calculated in WCAP–15049 for the accumulator CT increase meet the criterion of 5E–07 in Regulatory Guides (RG) 1.174 [''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis''] and 1.177 [''An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications''] for all cases except those that are based on design basis success criteria. As indicated in WCAP–15049, design basis accumulator success criteria are not considered necessary to mitigate large break loss-of-coolant accident (LOCA) events, and were only included in the WCAP–15049 evaluation as a worst case data point. In addition, WCAP–15049 states that the NRC has indicated that an incremental conditional core damage frequency (ICCDP) greater than 5E–07 does not necessarily mean the change is unacceptable.

The proposed technical specification change does not involve any hardware changes nor does it affect the probability of any event initiators. There will be no change to normal plant operating parameters, engineered safety feature (ESF) actuation setpoints, accident mitigation capabilities, accident analysis assumptions or inputs.

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

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. As described in Section 9.1 of the WCAP–15049 evaluation, the plant design will not be changed with this proposed technical specification CT increase. All safety systems still function in the same manner and there is no additional reliance on additional systems or procedures. The proposed accumulator CT increase has a very small impact on core damage frequency. The WCAP–15049 evaluation demonstrates that the small increase in risk due to increasing the accumulator allowed outage time (AOT) is within the acceptance criteria provided in RGs 1.174 and 1.177. No new accidents or transients can be introduced with the requested change and the likelihood of an accident or transient is not impacted.

The malfunction of safety related equipment, assumed to be operable in the accident analyses, would not be caused as a result of the proposed technical specification change. No new failure mode has been created and no new equipment performance burdens are imposed.

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

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

The proposed change does not involve a significant reduction in a margin of safety. There will be no change to the departure from nucleate boiling ratio (DNBR) correlation limit, the design DNBR limits, or the safety analysis DNBR limits.

The basis for the accumulator LCO, as discussed in Bases Section 3.5.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of the WCAP–15049 evaluation, the proposed change will allow plant operation with an inoperable accumulator for up to 24 hours, instead of 1 hour, before being required to begin shutdown. The impact of this on plant risk was evaluated and found to be very small. That is, increasing the time the accumulators will be unavailable to respond to a large LOCA event, assuming accumulators are needed to mitigate the design basis event, has a very small impact on plant risk. Since the frequency of a design basis large LOCA (a large LOCA with loss of offsite power) would be significantly lower than the large LOCA frequency of the WCAP–15049 evaluation, the impact of increasing the accumulator CT from 1 hour to 24 hours on plant risk due to a design basis large LOCA would be significantly less than the plant risk increase presented in the WCAP–15049 evaluation.

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

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

Attorney for licensee: Daniel F. Stenger, Ballard Spahr Andrews & Ingersoll, LLP, 601 13th Street, NW., Suite 1000 South, Washington, DC 20005.

NRC Section Chief: Richard J. Laufer.

Rochester Gas and Electric Corporation, Docket No. 50–244, R.E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: March 1, 2004.

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

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–359. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 2, 2002 (67 FR 50475), on possible amendments concerning TSTF–359, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on April 4, 2003 (68 FR 16579). The licensee affirmed the applicability of the following NSHC determination in its application dated March 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 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: Daniel F. Stenger, Ballard Spahr Andrews & Ingersoll, LLP, 601 13th Street, NW., Suite 1000 South, Washington, DC 20005.

NRC Section Chief: Richard J. Laufer.

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: April 28, 2004.

Description of amendment request: The proposed amendments would relocate requirements related to the Cold Over Pressure Protection System (COPS) arming temperature from the Technical Specifications (TSs) to the Pressure and Temperature Limits Report (PTLR) to facilitate future licensee-controlled changes to the COPS arming temperature. The licensee also proposed to change the COPS arming temperature from 350 °F to 220 °F.

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

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

No. The proposed changes to the Technical Specifications do not affect any plant equipment, test methods, or plant operation, and are not initiators of any analyzed accident sequence. COPS will continue to perform its function as designed to provide cold over pressure protection, and the pressurizer safety valves will provide over pressure protection during operation when COPS is not in service. Operation in accordance with the proposed TS will ensure that all analyzed accidents will continue to be mitigated by the Structures, Systems, and Components (SSCs) as previously analyzed. 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?

No. The proposed changes do not introduce any new equipment, create new failure modes for existing equipment, or create any new limiting single failures. COPS will continue to ensure that appropriate fracture toughness margins are maintained to protect against reactor vessel failure during low temperature operation. The proposed changes are consistent with [technical specification task force] TSTF–233, Revision 0, which was approved by the NRC. Plant operation will not be altered, and all safety functions will continue to perform as previously assumed in accident analyses. 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?

No. The proposed changes will not adversely affect the operation of plant equipment or the function of any equipment assumed in the accident analysis. The COPS arming temperature has been established in accordance with an NRC-approved methodology. No changes are being made to the cold Over pressure protection analysis and the function of COPS as assumed in the analysis. Therefore, the proposed changes do not involve a significant reduction in any margin to safety.

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

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

NRC Section Chief: Stephanie M. Coffin, Acting Section Chief.

Yankee Atomic Electric Co., Docket No. 50–29, Yankee Nuclear Power Station (YNPS) Franklin County, Massachusetts

Date of amendment request: November 24, 2003, and supplemented December 10, 2003, December 16, 2003, January 19, 2004, January 20, 2004, February 2, 2004, February 10, 2004, and March 4, 2004.

Description of amendment request: The licensee has proposed to amend its license to incorporate a new license condition addressing the license termination plan (LTP). The new license condition would document the date of NRC approval of the LTP and provide criteria to determine the need for NRC approval of changes to the approved LTP.

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

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

Currently, the bounding airborne radioactivity event given in the YNPS [Yankee Nuclear Power Station] FSAR [Final Safety Analysis Report] is the materials handling event (FSAR Section 403.5). This event considered the non-mechanistic release of the contents of the dominant plant component that could have caused the highest offsite dose as a result of the release of airborne radioactivity during handling. The dominant component was the feed and bleed heat exchanger which has since been removed from the site. The bounding analysis resulted in an offsite dose at the Exclusion Area Boundary of about 0.320 rem, significantly less than the EPA Protective Action Guidelines. Other airborne particulate radwaste or radioactive materials accidents considered in the FSAR but bounded by the materials handling event are as follows:

- Fire in a sea-land container containing combustible radioactive material,
- Dismantlement activities (*i.e.*, cutting , segmentation) during decommissioning,
- A gas bottle explosion inside containment,
- An explosion of a propane tank stored onsite.

All spent fuel is located at the ISFSI [Independent Spent Fuel Storage Installation] and is stored within fifteen NAC Multi-Purpose Canisters and associated vertical concrete casks. A sixteenth cask contains Greater Than Class C material. The NAC-MPC FSAR addresses the various off-normal and accident events which were postulated in support of the licensing and certification of the system. In each case, there were no radiological consequences as a result of a postulated event.

The requested license amendment is consistent with plant activities described in the PSDAR [Post Shutdown Decommissioning Activities Report] and the YNPS FSAR. Accordingly, no systems, structures, or components that could initiate the previously evaluated accident or are required to mitigate these accidents are adversely affected by this proposed change. Therefore, the proposed change does not involve an increase in the probability or consequences of any previously evaluated accident.

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

Accident analyses related to decommissioning activities are addressed in the FSAR. The requested license amendment is consistent with the plant activities described in the YNPS FSAR and the PSDAR. The proposed change does not affect plant systems, structures, or components in a way not previously evaluated. The changes do not affect any of the parameters or condition that could contribute to the initiation of an accident. No new accident scenarios are created nor are any new failure mechanisms created by this activity. Therefore, the proposed activity does not create the possibility of a new or different kind of accident than those previously evaluated.

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

The LTP [License Termination Plan] is a plan for demonstrating compliance with the radiological criteria for license termination as provided in 10 CFR 20.1402. The margin of safety defined in the statements of consideration for the final rule on the Radiological Criteria for License Termination is described as the margin between the 100 mrem/yr public dose limit established in 10 CFR 20.1301 for licensed operation and the 25 mrem/yr dose limit to the average member of the critical group at a site considered acceptable for unrestricted use (one of the criteria of 10 CFR 20.1402). This margin of safety accounts for the potential effect of multiple sources of radiation exposure to the critical group. Since the License Termination Plan was designed to comply with the radiological criteria for license termination for unrestricted use, the LTP supports this margin of safety.

In addition, the LTP provides the methodologies and criteria that will be used to perform remediation activities of residual radioactivity to demonstrate compliance with the ALARA [As Low As Reasonably Achievable] criterion of 10 CFR 20.1402.

Also, as previously discussed, the bounding accident for decommissioning is the materials handling event. Since the bounding decommissioning accident results in more airborne radioactivity than can be released from other decommissioning events, the margin of safety associated with the consequences of decommissioning accidents is not reduced by this activity. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Gerald Garfield, Esq., Day, Berry & Howard, City Place 1, Hartford, CT 06103.

NRC Section Chief: Claudia Craig.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois, Docket No. 50–219, Oyster Creek Generating Station, Ocean County, New Jersey, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania, Docket No. 50–289

Date of application for amendments: January 30, 2004.

Brief description of amendment: The amendments conformed the Operating Licenses to reflect the current ownership structure of AmerGen Energy Company, LLC. Exelon Generation Company currently owns 100% of AmerGen both directly and indirectly as a result of its purchase on December 22, 2003, of the stock of British Energy U.S. Holdings, Inc. The amendments deleted the License Conditions that are no longer valid as a result of the change of the AmerGen ownership.

Date of Issuance: May 27, 2004.

Effective date: These license amendments are effective as of their date of issuance.

Amendment Nos.: 160, 243, 249.

Facility Operating License Nos. DPR–16, DPR–50, and NPF–62: Amendments revised the Operating Licenses.

Date of initial notice in ***Federal Register:*** March 2, 2004 (69 FR 9859).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated May 27, 2004.

No significant hazards consideration comments received: No.

Calvert Cliffs Nuclear Power Plant, Inc., Docket No. 50–317, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Calvert County, Maryland

Date of application for amendment: May 1, 2003, as supplemented September 25, 2003, November 3, 2003, and February 25, 2004.

Brief description of amendment: The amendment adds Technical Specification (TS) 3.7.16, ''Spent Fuel Pool Boron Concentration,'' modifies TS 4.3.1, ''Criticality'' and adds an additional license condition that requires the licensee to develop a long-term coupon surveillance program for the Carborundum samples.

Date of issuance: June 3, 2004.

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

Amendment No.: 267.

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

Date of initial notice in ***Federal Register:*** May 27, 2003 (68 FR 28846).

The September 25, 2003, November 3, 2003, and February 25, 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 Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 3, 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: October 16, 2001; as supplemented by letters dated May 20, September 12, and November 21, 2002; September 22 and November 20, 2003; and February 18 and April 14, 2004.

Brief description of amendments: The amendments revised the Technical Specifications to incorporate changes resulting from use of an alternate source term.

Date of Issuance: June 1, 2004.

Effective date: These license amendments are effective as of the date of issuance and shall be implemented in accordance with the schedule provided in the licensee's letter dated February 18, 2004.

Amendment Nos.: 338, 339 & 339.

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

Date of initial notice in ***Federal Register:*** January 22, 2002 (67 FR 2922).

The supplements dated May 20, September 12, and November 21, 2002; and February 18 and April 14, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the **Federal Register** on January 22, 2002 (67 FR 2922). The supplements dated September 22, 2003, and November 20, 2003, did change the NRC staff's proposed no significant hazards consideration determination. The NRC staff's proposed no significant hazards consideration determination based on the submittals dated September 22, 2003, and November 20, 2003, were published in the **Federal Register** on October 14, 2003 (68 FR 59215), and December 9, 2003 (68 FR 68660), respectively.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 1, 2004.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of application for amendments: July 29, 2003, and as supplemented by submittal dated January 14, 2004.

Brief description of amendments: Revise the technical specifications by adding required actions for inoperable 250 VDC or 125 VDC battery charger, by relocating certain DC power surveillance requirements and criteria to a licensee controlled program, and by providing alternative criteria for battery charger testing and battery monitoring with required actions. Additionally, a new program for battery monitoring and maintenance is added to the technical specifications.

Date of issuance: June 8, 2004.

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

Amendment Nos.: 207/199.

Facility Operating License Nos. DPR–19 and DPR–25: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** October 14, 2003 (68 FR 59215).

The supplemental submittal contained clarifying information that was within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 8, 2004.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit No. 2, Beaver County, Pennsylvania

Date of application for amendment: February 4, 2003, as supplemented by letters dated October 24, 2003, and April 6, 2004.

Brief description of amendment: The amendment allowed the engineered safeguards features actuation system slave relay test frequency in footnote (1) to Technical Specification (TS) 4.3.2.1.1 to be changed from once per 92 days to once per 12 months provided a satisfactory contact loading analysis has been completed, and a satisfactory slave relay service life has been established, for the slave relay being tested.

Date of issuance: May 14, 2004.

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

Amendment No: 141.

Facility Operating License No. NPF–73. Amendment revised the TSs.

Date of initial notice in ***Federal Register:*** March 18, 2003 (68 FR 12953).

The supplements dated October 24, 2003, and April 6, 2004, 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 Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 14, 2004.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of application for amendment: January 30, 2004.

Brief description of amendment: The amendment eliminates requirements for hydrogen recombiners and relocates the requirements for hydrogen and oxygen monitors to the licensee's Commitment Tracking Program.

Date of issuance: May 21, 2004.

Effective date: As of the date of issuance and shall be implemented within 120 days.

Amendment No.: 138.

Facility Operating License No. DPR–22. Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 2, 2004 (69 FR 9862).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 21, 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 30, 2004, supplemented by letter dated May 6, 2004.

Brief description of amendments: The amendments eliminate requirements for hydrogen recombiners and relocate the requirements for hydrogen monitors to the Technical Requirements Manual.

Date of issuance: June 8, 2004.

Effective date: As of the date of issuance and shall be implemented within 120 days.

Amendment Nos.: 163 and 154.

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

Date of initial notice in ***Federal Register:*** March 2, 2004 (69 FR 9862).

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

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 8, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 1, 2003, as supplemented on March 10 and 30, 2004.

Brief description of amendments: The amendments revised the Technical Specifications to change the peak calculated post accident primary containment internal pressure values for the primary containment leakage rate testing program.

Date of issuance: May 28, 2004.

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

Amendment Nos.: 241 and 184.

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

Date of initial notice in ***Federal Register:*** January 20, 2004 (69 FR 2747).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 30, 2003.

Brief description of amendments: The amendments revised the staff position titles in Section 5.0 ''Administrative Controls'' of the Technical Specifications.

Date of issuance: June 3, 2004.

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

Amendment Nos.: 242 and 185.

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

Date of initial notice in ***Federal Register:*** March 2, 2004 (69 FR 9865).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 3, 2004.

No significant hazards consideration comments received: No.

STP Nuclear Operating Company, Docket No. 50–498, South Texas Project, Unit 1, Matagorda County, Texas

Date of amendment request: October 16, 2003, as supplemented March 3, 2004.

Brief description of amendments: The amendment provides a one-time change to Technical Specification 4.4.5.3a to extend the steam generator inspection interval to 44 months for STP, Unit 1.

Date of issuance: June 8, 2004.

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

Amendment No.: Unit 1—162.

Facility Operating License No. NPF–76: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 12, 2003 (68 FR 64139). The supplement dated March 4, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 8, 2004.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket No. 50–339, North Anna Power Station, Unit 2, Louisa County, Virginia

Date of application for amendment: January 23, 2004.

Brief description of amendment: This amendment revises Technical Specification Surveillance Requirements 3.5.1.4, 3.5.4.3, and 3.6.7.3 in order to delete a note that differentiates between the boron concentrations at North Anna, Units 1 and 2, for the safety injection accumulators, the refueling water storage tank, and the casing cooling tank.

Date of issuance: June 4, 2004.

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

Amendment No: 218.

Renewed Facility Operating License No. NPF–7: Amendment changes the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 30, 2004 (69 FR 16624).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 4, 2004.

No significant hazards consideration comments received: No.

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

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

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

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

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

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

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. Within 60 days after the date of publication of this notice, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, and electronically on the Internet at the NRC Web site, *http://www@nrc.gov/reading-rm/doc-collections/cfr/.* If there are problems in accessing the document, contact the PDR Reference staff at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner/requestor shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. The petition must include sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact.[1] Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner/requestor who fails to satisfy these requirements with respect to at least one contention will not be permitted to participate as a party.

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

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed by: (1) First class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff; (2) courier, express mail, and expedited delivery services: Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff; (3) e-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, *hearingdocket@nrc.gov;* or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC, Attention: Rulemakings and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

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

Duke Energy Corporation, Docket No. 50–270, Oconee Nuclear Station, Unit 2, Oconee County, South Carolina

Date of amendment request: June 4, 2004.

Description of amendment request: The amendment revised Technical Specification 3.6.5, ''Reactor Building Spray and Cooling Systems,'' to add a note that states that Limiting Condition of Operation 3.0.4 is not applicable.

Date of issuance: June 4, 2004.

Effective date: June 4, 2004.

Amendment No.: 340.

Facility Operating License No. DPR–47: Amendment revises the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): No. The Commission's related evaluation of the amendment, finding of emergency circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated June 4, 2004.

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

NRC Section Chief: Stephanie M. Coffin, Acting.

Dated at Rockville, Maryland, this 14th day of June 2004.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. 04–13753 Filed 6–21–04; 8:45 am]

BILLING CODE 7590–01–P

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

OVERSEAS PRIVATE INVESTMENT CORPORATION

Submission for OMB Review; Comment Request

AGENCY: Overseas Private Investment Corporation (OPIC).

ACTION: Request for comments.

SUMMARY: Under the provisions of the Paperwork Reduction Act (44 U.S.C. Chapter 35), agencies are required to publish a Notice in the **Federal Register** notifying the public that the Agency is preparing an information request for OMB review and approval and to request public review and comment on the submission. Comments are being solicited on the need for the information, the accuracy of the Agency's burden estimate; the quality, practical utility and clarity of the information to be collected; and on ways to minimize the reporting burden, including automated collection techniques and uses of other forms of technology. The proposed form, OMB control number 3420–0004, under review is summarized below.

DATES: Comments must be received within 60 calendar days of publication of this Notice.

ADDRESSES: Copies of the subject form and the request for review prepared for submission to OMB may be obtained from the Agency Submitting Officer. Comments on the form should be submitted to the Agency Submitting Officer.

FOR FURTHER INFORMATION CONTACT: