Persons who have an interest in reviewing these documents should submit a request to NRC under the Freedom of Information Act (FOIA). Instructions for submitting a FOIA request can be found on the NRC's Web site at *http://www.nrc.gov/reading-rm/foia/foia-privacy.html.*

Dated in King of Prussia, Pennsylvania this 31st day of May, 2005.

For the Nuclear Regulatory Commission.

James P. Dwyer,

Chief, Commercial and R&D Branch, Division of Nuclear Materials Safety, Region I.

[FR Doc. 05–11217 Filed 6–6–05; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETINGS: Nuclear Regulatory Commission.

DATE: Weeks of June 6, 13, 20, 27, July 4, 11, 2005.

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

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of June 6, 2005

There are no meetings scheduled for the week of June 6, 2005.

Week of June 13, 2005—Tentative

There are no meetings scheduled for the week of June 13, 2005.

Week of June 20, 2005—Tentative

There are no meetings scheduled for the week of June 20, 2005.

Week of June 27, 2005—Tentative

Tuesday, June 28, 2005.

9:30 a.m. Briefing on Equal Employment Opportunity (EEO) Program (Public Meeting) (Contact: Corenthis Kelley, 301–415–7380).

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

Wednesday, June 29, 2005.

9:30 a.m. Discussion of Security Issues (Closed—Ex. 1).

Week of July 4, 2005—Tentative

There are no meetings scheduled for the week of July 4, 2005.

Week of July 11, 2005—Tentative

There are no meetings scheduled for the week of July 11, 2005.

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

* * * * *

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

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The NRC provides reasonable accommodation to individuals with disabilities where appropriate. If you need a reasonable accommodation to participate in these public meetings, or need this meeting notice or the transcript or other information from the public meetings in another format (e.g. braille, large print), please notify the NRC's Disability Program Coordinator, August Spector, at 301–415–7080, TDD: 301–415–2100, or by e-mail at *aks@nrc.gov.* Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

* * * * *

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

Dated: June 2, 2005.

Dave Gamberoni,

Office of the Secretary.

[FR Doc. 05–11350 Filed 6–3–05; 9:41 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

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 13, 2005 to May 25, 2005. The last biweekly notice was published on May 24, 2005 (70 FR 29785).

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 ADAMS Public Electronic Reading Room on the Internet at the NRC Web site, *http://*

www.nrc.gov/reading-rm/adams.html. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of amendment request: May 18, 2005.

Description of amendment request: The proposed amendment would revise Fermi 2 Technical Specifications (TSs) to add Actions to Limiting Condition for Operation (LCO) 3.8.1, ''AC Sources—Operating,'' for one offsite circuit inoperable, for two offsite circuits inoperable, and for one offsite circuit and one or both emergency diesel generators (EDGs) in one Division inoperable, in accordance with Regulatory Guide 1.93, ''Availability of Electric Power Sources.'' The current Fermi 2 TSs contain only a single Action for one or two offsite circuits inoperable.

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

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

The proposed change to replace the existing LCO 3.8.1 Action C for one or two offsite circuits inoperable with a required Completion Time of 12 hours to be in MODE 3, and 36 hours to be in MODE 4, with new Actions C, D, and E to allow a single offsite circuit to be inoperable for up to 72 hours, two offsite circuits to be inoperable for up to 24 hours, and one offsite circuit and one or both EDGs in one Division to be inoperable for up to 12 hours, provided other Required Actions are taken is consistent with the NUREG 1433, ''Standard Technical Specifications General Electric Plants, BWR/4,'' criteria, and with the guidelines in Regulatory Guide 1.93. There is no change in plant design, and [Title 10 of the Code of Federal Regulations (10 CFR)] 10 CFR 50, Appendix A, General Design Criteria 17, ''Electric Power Systems'' will continue to be met. Increasing the Completion Times for inoperable offsite circuits will not significantly increase the potential for a loss of offsite power. This is due to the redundancy and diversity of the offsite electrical configuration at Fermi 2. Inoperability of an offsite circuit does slightly increase the potential for a loss of divisional power. The probability of losing the opposite division of offsite power in this condition is extremely small due to the physical separation of the offsite power sources that feed Fermi 2. Furthermore, the 10 CFR 50.65(a)(4) program monitors the condition of the offsite electrical system and switchyard configuration for each entry into the extended completion time to ensure that there is no significant increase in the probability or consequences of an accident.

The proposed change does not alter the operation of any plant equipment assumed to function in response to an analyzed event or otherwise increase its failure probability. Therefore, this change does not involve a significant increase in the probability or the consequences of any 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 does not alter the design, configuration, or method of operation of the plant. It simply provides longer Completion Times for inoperable offsite circuits. No physical or operational changes to the components of the A. C. power systems are being made by this change; therefore, no new system interactions are being created. The proposed change does not produce any parameters or conditions that could contribute to the initiation of accidents different from those already evaluated. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will replace the existing LCO 3.8.1 Action C for one or two offsite circuits inoperable with a required Completion Time of 12 hours to be in MODE 3, and 36 hours to be in MODE 4, with new Actions C, D, and E to allow a single offsite circuit to be inoperable for up to 72 hours, two offsite circuits to be inoperable for up to 24 hours, and one offsite circuit and one or both EDGs in one Division to be inoperable for up to 12 hours, provided other Required Actions are taken. This change is consistent with NUREG 1433, ''Standard Technical Specifications General Electric Plants, BWR/4,'' and with the guidelines in Regulatory Guide 1.93. The proposed change does not affect any analysis that is used to establish safety margins, nor does it alter the design, configuration, or method of operation of the plant. 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: David G. Pettinari, 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: April 19, 2005.

Description of amendment request: The proposed amendment would revise technical specifications (TS) testing frequency for the surveillance requirement (SR) in TS 3.1.4, ''Control Rod Scram Times.'' Specifically, the proposed change would revise the frequency for SR 3.1.4.2, Control Rod Scram Time Testing, from ''120 days cumulative operation in MODE 1'' to ''200 days cumulative operation in MODE 1.''

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in licensing amendment applications in the **Federal Register** on August 23, 2004 (69 FR 51864). The licensee affirmed the applicability of the model NSHC determination in its application dated April 19, 2005.

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:

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

Response: No.

The proposed change extends the frequency for testing control rod scram time testing from every 120 days of cumulative Mode 1 operation to 200 days of cumulative Mode 1 operation. The frequency of surveillance testing is not an initiator of any accident previously evaluated. The frequency of surveillance testing does not affect the ability to mitigate any accident previously evaluated, as the tested component is still required to be operable. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change extends the frequency for testing control rod scram time testing from every 120 days of cumulative Mode 1 operation to 200 days of cumulative Mode 1 operation. The proposed change does not result in any new or different modes of plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed change extends the frequency for testing control rod scram time testing from every 120 days of cumulative Mode 1 operation to 200 days of cumulative Mode 1 operation. The proposed change continues to test the control rod scram time to ensure the assumptions in the safety analysis are protected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, the proposed change 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.

Attorney for licensee: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

Entergy Nuclear Operations, Docket Nos. 50–247 and 50–286, Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and 3), Westchester County, New York

Date of amendment request: April 22, 2005.

Description of amendment request: The amendments would revise the surveillance requirements (SRs) for Technical Specification (TS) 3.3.5, ''Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation.'' Specifically, a note would be added to IP2 TS SR 3.3.5.2 to indicate that the verification of the setpoint is not required for the 480 volt (V) bus degraded voltage function when performing the trip actuating device operational test (TADOT). A similar note would be added to IP3 TS SR 3.3.5.1 for the 480V degraded voltage and undervoltage 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. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated[?]

Response: No.

The proposed change adds a note to indicate that the IP2 and IP3 degraded voltage relays and the IP3 undervoltage relays do not require setpoint verification when the TADOT required by TS surveillances is performed on a monthly basis. Setpoint verification of these relays occurs as part of the channel calibration that is performed at either an 18 month or a 24 month frequency. These relays are used to sense either degraded voltage or undervoltage on the 480 volt safety related buses and to initiate the start of the EDG [emergency diesel generator] for all events where the loss of offsite power is postulated. This function has no effect on the probability of an accident previously evaluated since it is not associated with the initiation of any accident. The relay setpoint verification frequency of 18 or 24 months has no significant effect on the consequences of an accident because the relays are intended to be calibrated on this frequency. This frequency of calibration is based on operating experience, and is consistent with industry practice. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change adds a note to indicate that the IP2 and IP3 degraded voltage relays and the IP3 undervoltage relays do not require setpoint verification when the TADOT required by TS surveillances is performed on a monthly basis. This effectively changes the frequency required by the surveillance requirement from 31 days to either 18 months or 24 months. The change does not affect the function of the relays or otherwise affect the design and operation of plant systems and components and therefore no new accident scenarios would be created. The change does not affect the manner is which equipment is operated but does affect the manner in which it is maintained by extending the frequency for setpoint verification. The frequency change continues to provide adequate verification of the operability of equipment and limits the time which the relay function is inoperable or degraded while performing verification. Therefore, no new failure modes are being introduced that could lead to different accidents.

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

Response: No.

The proposed change adds a note to indicate that the IP2 and IP3 degraded voltage relays and the IP3 undervoltage relays do not require setpoint verification when the TADOT required by TS surveillances is performed on a monthly basis. Setpoint verification of these relays occurs as part of the channel calibration that is performed at either an 18 month or a 24 month frequency. The margin associated with these relays is the assurance that these relays will properly sense either degraded voltage or undervoltage on the 480 volt safety related buses and to initiate the start of the EDG for all events where the loss of offsite power is postulated. The proposed frequency of calibration is based on operating experience, and is consistent with industry practice. These indicate that setpoint verification at 18 month or 24 month [frequency] is adequate to assure performance of the function. Verification of setpoints on a monthly basis either degrades the reliability of the function or makes it inoperable. Therefore, the proposed change does not involve a significant reduction in [a] margin of safety.

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

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: April 13, 2005.

Description of amendment request: The proposed amendments would extend the completion time (CT) for required Action A.1, ''Restore Residual Heat Removal Service Water (RHRSW) subsystem to OPERABLE status,'' associated with Technical Specification (TS) Section 3.7.1 from 7 days to 10 days. This proposed change would only be used during the upcoming Unit 1 2006 refueling outage. The establishment of a 6 day (for Division 2 core standby cooling system (CSCS) maintenance) or 10 day (for Division 1 CSCS maintenance) CT for TS Section 3.7.2 when one or more required diesel generator cooling water (DGCW) subsystem(s) are inoperable. This proposed change will only be used during each of the upcoming Unit 1 2006, and Unit 2 2007, refueling outages, and during the subsequent Unit 1 2008, refueling outage. An extension of the CT for required Action C.4, ''Restore required Diesel Generator (DG) to OPERABLE status,'' associated with TS Section 3.8.1 from 72 hours to 6 days. This proposed change will only be used during the upcoming Unit 2 2007 refueling outage, and during subsequent Unit 1, 2008, refueling outage. An extension of the CT for required Action F.1, ''Restore one required Diesel Generator (DG) to OPERABLE status,'' associated with TS Section 3.8.1 from 2 hours to 6 days. This proposed change will only be used during the upcoming Unit 2, 2007, refueling outage, and during subsequent Unit 1, 2008, refueling outage.

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

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

The proposed changes have been evaluated using the risk-informed processes described in RG [Regulatory Guide] 1.174, ''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,'' dated July 1998, and RG 1.177, ''An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications,'' dated August 1998. The risk associated with the proposed change was found to be acceptable.

The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed change does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. No active or passive failure mechanisms that could lead to an accident are affected. Non-code line stops required to isolate the Unit 1 portion of the common discharge header from the Unit 2 portion of the header during the specified CSCS maintenance will maintain the availability of the online unit's Division 2 CSCS system. The non-code line stops being used to isolate the system during the specified refueling outages are being designed to the same pressure rating and seismic requirements as the CSCS piping.

Redundancy is provided by designing the CSCS system as multiple independent subsystems. Separation between subsystems assures that no single failure can affect more than one subsystem. Therefore, assuming a single failure in any subsystem including the subsystem shared between units, two subsystems in each unit will remain unaffected. These two subsystems can supply the minimum required cooling water for safe shutdown of a unit or mitigate the consequences of an accident.

The proposed limited use of increased CT's of the operating unit's CSCS system maintains the design basis assumptions; therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change involves the temporary installation of new equipment (mechanical line stops) that will be designed and installed to the same pressure rating and seismic design as the CSCS piping. The currently installed equipment will not be operated in a new or different manner. No new or different system interactions are created and no new processes are introduced. The proposed changes will not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases. Based on this evaluation, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not alter any existing setpoints at which protective actions are initiated and no new setpoints or protective actions are introduced. The design and operation of the CSCS system remains unchanged. The risk assessment with the proposed increase in the CTs for TS 3.7.1, TS 3.7.2, and TS 3.8.1 were evaluated using the risk-informed processes described in RG 1.174, ''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,'' dated July 1998, and RG 1.177, ''An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications,'' dated August 1998. The risk was shown to be acceptable. Based on this evaluation, the proposed change does not involve a significant reduction in a margin of safety.

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

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

Date of amendment request: April 11, 2005.

Description of amendment request: The proposed amendment would revise the BVPS–2 Technical Specification (TS) 3.4.5 to change the scope of the steam generator (SG) tubesheet examinations required in the SG tubesheet region by using the F* inspection methodology. Specifically, the proposed amendment would alter the tube inspection to exclude the portion of the SG tube within the tubesheet below the F* distance and to exclude the tube-to-tubesheet weld, by crediting the methodology described in Westinghouse Topical Report, WCAP–16385, Revision 1. The F* distance is the distance from the top of the tubesheet to the bottom of the F* length (the maximum length of tubing below the bottom of the roll transition (BRT) which must be demonstrated to be non-degraded and which is defined as 1.97 inches on the hot leg side) plus the distance to the BRT and non-destructive examination uncertainties. The licensee's proposed amendment also would revise the TS requirements to require tubes with service-induced degradation identified in the F* distance or less than or equal to 3.0 inches below the top of the tubesheet, whichever is greater, to be repaired or removed from service upon detection. The TS Index, affected TS pages and Bases would also be revised and repaginated as necessary to reflect the proposed TS change.

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

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

No. The proposed change modifies the BVPS Unit 2 TSs to incorporate steam generator tube inspection scope based on WCAP–16385, Revision 1. Of the various accidents previously evaluated in the BVPS Unit 2 Updated Final Safety Analysis Report (UFSAR), the proposed changes only affect the steam generator tube rupture (SGTR) event evaluation and the postulated steam line break (SLB) accident evaluation. Loss-of-coolant accident (LOCA) conditions cause a compressive axial load to act on the tube. Therefore, since the LOCA tends to force the tube into the tubesheet rather than pull it out, it is not a factor in this amendment request. Another faulted load consideration is a safe shutdown earthquake (SSE); however, the seismic analysis of Model 51M SGs has shown that axial loading of the tubes is negligible during an SSE.

For the SGTR event, the required structural margins of the steam generator tubes will be maintained by the presence of the tubesheet. Tube rupture is precluded for cracks in the tube expansion region due to the constraint provided by the tubesheet. Therefore, Regulatory Guide (RG) 1.121, ''Bases for Plugging Degraded PWR [pressurized-water reactor] Steam Generator Tubes,'' margins against burst are maintained for both normal and postulated accident conditions.

The F* length supplies the necessary resistive force to preclude pullout loads under both normal operating and accident conditions. The contact pressure results from the tube expansion process used during manufacturing and from the differential pressure between the primary and secondary side. The proposed changes do not affect other systems, structures, components or operational features. Therefore, the proposed change results in no significant increase in the probability of the occurrence of an SGTR or SLB accident.

The consequences of an SGTR event are affected by the primary-to-secondary leakage flow during the event. Primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed change since the tubesheet enhances the tube integrity in the region of the expansion by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and collapse is strengthened by the tubesheet in that region. At normal operating pressures, leakage from primary water stress corrosion cracking (PWSCC) below the F* length is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region.

SLB leakage is limited by leakage flow restrictions resulting from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of crack face opening compared to free span indications. The total leakage (i.e., the combined leakage for all such tubes) meets the industry performance criterion, plus the combined leakage developed by any other alternate repair criteria, and will be maintained below the maximum allowable SLB leak rate limit, such that off-site doses are maintained less than 10 CFR [Part] 100 guideline values and the limits evaluated in the BVPS Unit 2 UFSAR.

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

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

No. The proposed changes do not introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Tube bundle integrity will continue to be maintained for all plant conditions upon implementation of the F* methodology.

The proposed changes do not introduce any new equipment or any change to existing equipment. No new effects on existing equipment are created nor are any new malfunctions introduced.

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?

No. The proposed changes maintain the required structural margins of the steam generator tubes for both normal and accident conditions, including the planned uprated power level of 2910 Mwt. NRC [Nuclear Regulatory Commission] Regulatory Guide (RG) 1.121 is used as the basis in the development of the F* methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC staff for meeting General Design Criteria 14, 15, 31, and 32 by reducing the probability and consequences of an SGTR. RG 1.121 concludes that by determining the limiting safe conditions of tube wall degradation beyond which tubes with unacceptable cracking, as established by inservice inspection, should be removed from service or repaired, the probability and consequences of an SGTR are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code.

For primarily axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. WCAP–16385, Revision 1, defines a length, F*, of degradation-free expanded tubing that provides the necessary resistance to tube pullout due to the pressure-induced forces (with applicable safety factors applied). Application of the F* criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for large margins between calculated and actual leakage values in the F* criteria.

Plugging of the steam generator tubes reduces the reactor coolant flow margin for core cooling. Implementation of F* methodology at Beaver Valley Unit 2 will result in maintaining the margin of flow that may have otherwise been reduced by tube plugging.

Based on the above, it is concluded that the proposed changes do not result in a significant reduction of margin with respect to plant safety as defined in the Final Safety Analysis Report Update or bases of the plant Technical Specifications.

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

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

NRC Section Chief: Richard J. Laufer.

Florida Power and Light Company, Docket No. 50–389, St. Lucie Plant, Unit No. 2 (SL2), St. Lucie County, Florida

Date of amendment request: March 31, 2005.

Description of amendment request: The proposed amendment would revise Administrative Technical Specification Section 6.8.4.h, ''Containment Leakage Rate Testing Program,'' to allow a one-time extension of the currently approved 15-year test interval to approximately 15.5 years.

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) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed amendment of the Technical Specifications adds a one-time extension to the current surveillance interval for Type A testing (ILRT [integrated leak rate testing]). The current test interval of 15 years from the last Type A test would be extended to end prior to startup from the SL2–17 refueling. This is anticipated to be an approximately six-month addition to the 15 year interval. The proposed extension to the Type A testing interval does not significantly increase the probability of an accident previously evaluated since the containment Type A test is not a modification, nor a change in the way that plant systems, structures or components (SSC) are operated, and is not an activity that could lead to equipment failure or accident initiation. The proposed extension of the test interval does not involve a significant increase in the consequences of an accident since research documented in NUREG–1493 has found that generically, very few potential leak paths are not identified with Type B and C tests (LLRT [local leak-rate test]). The Type B and C testing are unaffected by this proposed change. The NUREG concluded that an increase in the Type A test interval to twenty years resulted in an imperceptible increase in risk. St. Lucie Unit 2 provides a high degree of assurance through testing and inspection that the containment will not degrade in a manner only detectable by Type A testing. Inspections required by the ASME [American Society of Mechanical Engineers] Code, the containment leakage rate testing program, the plant protective coatings program, and Maintenance Rule are performed in order to identify indications of containment degradation that could affect leak tightness. Type B and C testing required by 10 CFR 50, Appendix J, are not affected by this proposed extension to the Type A test interval and will identify openings in containment penetrations that would otherwise require a Type A test.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed change does not result in facility operation that would create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed extension to Type A testing does not create a new or different type of accident for St. Lucie because no physical plant changes are made and no compensatory measures are being imposed that could potentially lead to a failure. There are no operational changes that could introduce a new failure mode or create a new or different kind of accident. The proposed change only adds an extension to the current interval for Type A testing and does not change implementation aspects of the test.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The proposed change would not result in operation of the facility involving a significant reduction in a margin of safety. The proposed license amendment adds a one-time extension to the current interval for Type A testing (ILRT). The current one-time test interval of 15 years from the last Type A test would be extended to end prior to startup from the SL2–17 refueling outage. This is anticipated to be an approximately six month addition to the 15 year interval.

The NUREG–1493 generic study of the effects of extending the Type A test interval out to 20 years concluded that there is an imperceptible increase in plant risk. A plant specific risk calculation obtained results consistent with the generic conclusions regarding risk which show a slight but negligible increase in risk. Inspections required by the ASME code and maintenance rule are performed to ensure that the containment will not degrade in a manner that is only detectable by Type A testing (ILRT).

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

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

NRC Section Chief: Michael L. Marshall, Jr.

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

Date of amendment request: April 13, 2005.

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

1. TSTF–222–A, Revision 1, ''Control Rod Scram Time Testing''—This change modifies TS Section 3.1.4, ''Control Rod Scram Times,'' to clarify that control rod scram time testing is required only for core cells in which work on the control rod or drive has been performed or fuel has been moved or replaced.

2. TSTF–275–A, Revision 0, ''Clarify Requirement for EDG [emergency diesel generator] start signal on RPV [reactor pressure vessel] Level—Low, Low, Low during RPV cavity flood-up''—This change modifies the TS Section 3.3.5.1, ''ECCS [emergency core cooling system] Instrumentation,'' to clarify that the ECCS initiation instrumentation, identified as being required in modes 4 and 5, is required to be operable only when the associated ECCS subsystems are required to be operable as defined in limiting condition of operation (LCO) 3.5.2, ''ECCS—Shutdown.''

3. TSTF–300–A, Revision 0, ''Eliminate DG [diesel generator] LOCA [loss-of-coolant accident]—Start SRs [surveillance requirements] while in S/D [shutdown] when no ECCS is Required''—This change modifies the TS Section 3.8.2, ''AC [alternating current] Sources—Shutdown,'' to add an additional note to the surveillance that verifies automatic start of the emergency diesel generators and automatic load shedding from the emergency buses, is considered to be met without the ECCS initiation signals operable when ECCS initiation signals are not required to be operable per Table 3.3.5.1–1, ECCS Instrumentation.

4. TSTF–225, Revision 2, ''Fuel movement with inoperable refueling equipment interlocks''—This change modifies TS Section 3.9.1, ''Refueling Equipment Interlocks,'' to add required actions to allow insertion of a control rod withdrawal block and verification that all control rods are fully inserted as alternate actions to suspending in-vessel fuel movement in the event that one or more required refueling equipment interlocks are inoperable.

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

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

Response: No.

1. *Revision of CNS [Cooper Nuclear Station] TS SR 3.1.4.1 and SR 3.1.4.4.* The frequency at which control rod scram time is verified is not a precursor of an accident. A scram time slower than required might result in an increase in the consequences of an accident. However, revising the frequency for verifying the scram time of the control rods does not impact the scram time. Verifying that the scram time is acceptable will continue to be required prior to plant startup following fuel movement or work on the control rods or control rod drive system. Therefore, revising the frequency for verifying insertion time to clarify when it is required does not involve a significant increase in the probability of an accident or an increase in the consequences of an accident.

2. *Revision of TS Table 3.3.5.1–1.* Clarifying when certain ECCS instrumentation must be operable with the plant shut down will not increase either the probability of an accident or the consequences of the accident. The ECCS instrumentation is required to be operable only when the associated ECCS subsystems are required to be operable. This continues to ensure that the instrumentation will be operable when it is required.

3. *Revision of TS SR 3.8.2.1.* The frequency of verifying certain actions by surveillances is not a precursor to accidents. Clarifying that the actions required in response to an ECCS initiation signal are not required when the ECCS initiation signals are not required to be operable does not result in increased probability of an accident or increased consequences of an accident. Not requiring that a DG automatically start in response to the ECCS initiation signal when the ECCS subsystems that are supported by the DG are not required to be operable does not reduce the required ECCS protection.

4. *Revision of TS 3.9.1., Condition A Required Action.* The actions taken when a refueling equipment interlock is inoperable are not initiators of any accident previously evaluated. The level of protection against withdrawing a control rod during the insertion of a fuel assembly or loading a fuel assembly into the vessel with a control rod withdrawn, provided by the proposed alternate Required Actions, is equivalent to that provided by the current Required Action. The radiological consequences of an accident described in the Updated Safety Analysis Report (USAR) while taking the proposed alternate Required Actions are not different from the consequences of an accident under the current Required Actions.

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?

Response: No.

The proposed changes to the CNS operating license involve revisions to the requirements for when certain surveillances are to be performed (change no. 1 and no. 3), clarification of when ECCS instrumentation is required to be operable (change no. 2), and addition of alternative Required Actions if certain plant components are inoperable (change no. 4). These changes will not result in revision of plant design, physical alteration of a plant structure, system, or component (SSC), or installation of a new or different type of equipment. The changes do not involve any revision of how the plant, an SSC, or a refueling equipment interlock, are operated. Based on this, the proposed changes do not create the possibility of a new or different kind of accident.

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

Response: No.

1. *Revision of CNS TS SR 3.1.4.1 and SR 3.1.4.4.* Sufficiently rapid insertion of control rods following certain accidents (scram time) will prevent fuel damage, and thereby maintain a margin of safety to fuel damage. No change is being made to the required insertion rate specified in plant technical specifications. Clarifying when control rod insertion times must be verified following movement of fuel assemblies, without actually changing the requirement (verification of insertion times will continue to be required whenever work that might impact the rod insertion time is done), does not reduce the margin of safety related to fuel damage.

2. *Revision of TS Table 3.3.5.1–1.* Clarifying when certain ECCS instrumentation is required to be operable when CNS is in a shutdown mode does not change the requirement. Not requiring ECCS signals that initiate a DG to be operable when the ECCS subsystems that are supported by

the DG are not required to be operable does not result in a reduction of a margin of safety for the safety related equipment that is required to be operable.

3. *Revision of TS SR 3.8.2.1.* Clarifying that automatic start of the DGs in response to the ECCS initiation signal is not required when the ECCS subsystems that are supported by the DG are not required to be operable does not result in a reduction in a margin of safety.

4. *Revision of TS 3.9.1, Condition A Required Action.* The proposed alternate Required Actions to be taken when a refueling interlock is inoperable provide a level of protection against inadvertent criticality while inserting or moving fuel in the reactor vessel that is equivalent to the level provided by the current Required Action. As a result, the proposed alternate Required Actions do not result in a significant reduction in a margin of safety related to protection against inadvertent criticality when inserting or moving fuel assemblies.

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 C. McClure, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Section Chief: David Terao.

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

Date of amendment request: February 25, 2005.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.1.3.1, ''Control Rod Operability,'' such that scram discharge volume (SDV) vent or drain lines with inoperable valves would be isolated instead of requiring that the valve be restored to Operable status or the unit be placed in Hot Shutdown within 12 hours.

The NRC staff issued a Notice of Opportunity for Comment in the **Federal Register** on February 24, 2003 (68 FR 8637), on possible amendments to revise the action for one or more SDV vent or drain lines with an inoperable valve, 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 license amendment applications in the **Federal Register** on April 15, 2003 (68 FR 18294). The licensee affirmed the applicability of the model NSHC determination (modified slightly to address plant-specific TS format) in its application dated February 25, 2005.

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.

A change is proposed to allow the affected SDV vent and drain line to be isolated when there are one or more SDV vent or drain lines with inoperable valves instead or requiring the valves to be restored to operable status or the unit be in hot shutdown within 12 hours. With SDV vent or drain valves inoperable in one or more lines, the isolation function would be maintained since the redundant valve in the affected line would perform its safety function of isolating the SDV. Following the completion of the required action, the isolation function is fulfilled since the associated line is isolated. The ability to vent and drain the SDV is maintained and controlled through administrative controls. This requirement assures the reactor protection system is not adversely affected by the inoperable valves. With the safety functions of the valves being maintained, the probability or consequences of an accident previously evaluated are not significantly increased.

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

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

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

The proposed change ensures that the safety functions of the SDV vent and drain valves are fulfilled. The isolation function is maintained by redundant valves and by the required action to isolate the affected line. The ability to vent and drain the SDV is maintained through administrative controls. In addition, the reactor protection system will prevent filling of the SDV to the point that it has insufficient volume to accept a full scram. Maintaining the safety functions related to isolation of the SDV and insertion of control rods ensures that the proposed change does not involve a significant reduction in the margin of safety.

Based on the reasoning presented above, 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: Darrell J. Roberts.

R.E. Ginna Nuclear Power Plant, LLC, Docket No. 50–244, R.E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: March 10, 2005.

Description of amendment request: The amendment would revise Technical Specification Section 5.5.15, ''Containment Leakage Rate Testing Program,'' to allow a one-time extension of the interval between the Type A, integrated leakage rate tests (ILRTs), from 10 years to no more than 15 years.

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 to Technical Specification 5.5.15, Containment Leakage Rate Testing Program, involves a one-time extension to the current interval for Type A containment testing. The current test interval of ten (10) years would be extended on a one-time basis to no longer than fifteen (15) years from the last Type A test.

The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner which the plant is operated or controlled. The reactor containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such the reactor containment itself and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident.

The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications. Industry experience has shown, as documented in NUREG–1493, that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. The Ginna ILRT test history supports this conclusion. In NUREG–1493 Section 10, Summary of Technical Findings, it is concluded, in part, that reducing the frequency of Type A containment leak tests to once per twenty (20) years leads to an imperceptible increase in risk.

The proposed change does not result in an increase in core damage frequency since the containment system is used for mitigation purposes only. Containment Leakage Rate Testing Program local leak rate test requirements and administrative controls such as design change control, ASME [American Society of Mechanical Engineers] Section XI Inservice Inspection (ISI) Program Containment Repair and Replacement Program and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the reactor containment itself combined with the containment inspections performed in accordance with the ASME Section XI Inservice Inspection (ISI) Program Containment Program, Boric Acid Corrosion Program, inspections in accordance with Regulatory Guide 1.163 position C.3 and the Maintenance Rule serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing.

Therefore, the proposed Technical Specification change does not involve a significant increase in the 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 to Technical Specification 5.5.15 involves a one-time extension to the current interval for Type A containment testing. The reactor containment and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve the prevention or identification of any precursors of an accident. The proposed Technical Specification change does not involve a physical change to the plant (*i.e.*, no new or different type of equipment will be installed) or changes in the methods in which the plant is operated or controlled.

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

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

The proposed change to Technical Specifications involves a one-time extension to the current interval for Type A containment testing. The proposed Technical Specification change does not alter the manner in which safety limits, limiting safety system set points, or limiting conditions for operation are determined. The specific requirements and conditions of the Primary Containment Leakage Rate Testing Program, as defined in Technical Specifications, exist to ensure that the degree of reactor containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leakage rate limit specified by Technical Specifications is maintained. The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications.

Ginna and industry experience strongly supports the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with the ASME Section XI Inservice Inspection (ISI) Program Containment Program, Boric Acid Corrosion Program, inspections in accordance with Regulatory Guide 1.163 position C.3 and the Maintenance Rule serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. The combination of these factors ensures that the margin of safety that is inherent in plant safety analysis is maintained.

Therefore, the proposed Technical Specification 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.

R.E. Ginna Nuclear Power Plant, LLC, Docket No. 50–244, R.E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: April 29, 2005.

Description of amendment request: The amendment would revise Technical Specification Section 3.7.3, ''Main Feedwater Regulating Valves (MFRVs), Associated Bypass Valves, and Main Feedwater Pump Discharge Valves (MFPDVs),'' to allow the use of the main feedwater isolation valves in lieu of the main feedwater pump discharge valves to provide isolation capability to the steam generators in the event of a steam line break.

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:

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

Response: No.

The proposed changes involve a modification to the plant configuration to ensure the acceptability of containment response for Steam Line Breaks (SLB) inside containment.

The changes have also been evaluated to ensure the core response for steam system piping breaks remains acceptable. The changes to the Technical Specifications (TS) are necessary to properly accommodate the changes in plant configuration and ensure proper testing of the modified components.

The proposed changes do not adversely affect accident initiators or precursors nor significantly alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not adversely alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase the types and amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes cannot affect the probability of an accident occurring since they reflect a change in plant design consistent with current design which is not an accident initiator. The proposed changes cannot increase the consequences of postulated accidents since they reflect a change in plant design that will continue to mitigate the effects of feedwater addition to a faulted steam generator for a main steam line break inside containment.

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

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

Response: No.

The proposed changes involve a modification to the plant configuration to ensure the acceptability of containment response for Steam Line Breaks (SLB) inside containment. The changes have also been evaluated to ensure the core response for steam system piping breaks remains acceptable. The changes to the Technical Specifications (TS) are necessary to properly accommodate the changes in plant configuration and ensure proper testing of the modified components.

The change in plant configuration significantly reduces the available water volume and therefore the mass and energy released to the containment in the event of an SLB with failure of a feedwater regulating valve. Existing feedwater flow paths or piping are not significantly altered. An existing manual valve in the flow path to each steam generator is utilized as the main feedwater isolation valve by the addition of an air actuator to provide automatic isolation capability. The changes do not involve a significant change in the methods governing normal plant operation. The TS changes modify the limiting condition for operation, required action statements, associated completion times and surveillance requirements to those that are consistent with those previously approved for Westinghouse plants in the Standard Technical Specifications found in NUREG–1431. The proposed TS changes do not create the possibility of a new or different [kind] of accident from those previously evaluated since they reflect a design change that will accomplish the same feedwater isolation function as previously performed by the main feedwater pump discharge isolation valves with no significant change to the manner in which the feedwater system operates.

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

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

Response: No.

The proposed changes involve a modification to the plant configuration to ensure the acceptability of containment response for Steam Line Breaks (SLB) inside containment. The changes have also been evaluated to ensure the core response for steam system piping breaks remains acceptable. The changes to the Technical Specifications (TS) are necessary to properly accommodate the changes in plant configuration and ensure proper testing of the modified components.

The level of safety of facility operation is unaffected by the proposed changes since there is no change in the intent of the TS requirements of assuring proper main feedwater isolation in the event of a steam line break inside containment. The response of the plant systems to accidents and transients reported in the Updated Final Safety Analysis Report (UFSAR) is not adversely affected by this change. Therefore, the capability to satisfy accident analysis acceptance criteria is not adversely affected. The TS changes modify the limiting condition for operation, required action statements, associated completion times and surveillance requirements to those that are consistent with those previously approved for Westinghouse plants in the Standard Technical Specifications found in NUREG–1431. The proposed TS changes do not involve a significant reduction in [a] margin of safety since they are based upon a modification that will maintain [a] margin of safety with respect to feedwater addition for a main steam line break inside containment to the previously analyzed condition. Therefore, the 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: Daniel F. Stenger, Ballard Spahr Andrews & Ingersoll, LLP, 601 13th Street, NW., Suite 1000 South, Washington, DC 20005.

NRC Section Chief: Richard J. Laufer.

R.E. Ginna Nuclear Power Plant, LLC, Docket No. 50–244, R.E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: April 29, 2005.

Description of amendment request: The amendment would revise Technical Specification (TS) 3.5.1, ''Accumulators,'' and TS 3.5.4, ''Refueling Water Storage Tank (RWST),'' to reflect the results of revised analyses performed to accommodate a planned power uprate for the facility and revise TS 5.6.5, ''Core Operating Limits Report (COLR),'' to permit the use of NRC-approved methodology for large-break and small-break loss-of-coolant accidents (LOCAs).

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:

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

Response: No.

The proposed changes include revising accumulator volume and boron concentration requirements and Refueling Water Storage Tank (RWST) boron concentration requirements that are necessary to accommodate expected changes in the nuclear fuel (*e.g.*, higher enrichment) that are associated with the planned power uprate. Additionally, the change would allow Ginna to utilize analysis methodologies that have been previously approved for use at Westinghouse nuclear plants. The changes to the TS are necessary to ensure the acceptability of these systems to perform their intended function in the event of an accident.

The proposed changes do not adversely affect accident initiators or precursors nor significantly alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not adversely alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase the types and amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes cannot affect the probability of an accident occurring since they reflect a necessary change in plant design consistent with current design which is not an accident initiator. The proposed changes cannot increase the consequences of postulated accidents since they reflect a change in plant design that will continue to mitigate the effects of potential accidents. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed changes include revising accumulator volume and boron concentration requirements and RWST boron concentration requirements that are necessary to accommodate expected changes in the nuclear fuel (*e.g.*, higher enrichment) that are associated with the planned power uprate. Additionally, the change would allow Ginna to utilize analysis methodologies that have been previously approved for use at Westinghouse nuclear plants. The changes to the TS are necessary to ensure the acceptability of these systems to perform their intended function in the event of an accident.

The proposed changes involve changes to accumulator volume and boron concentration requirements and RWST boron concentration requirements to ensure the continued acceptability of LOCA and post LOCA analysis results. The changes to the Technical Specifications (TS) are necessary to properly accommodate the changes in plant design. The changes ensure applicable acceptance criteria will continue to be met. The changes do not involve a significant change in the methods governing normal plant operation. The proposed TS changes do not create the possibility of a new or different [kind] of accident from those previously evaluated since they reflect a change that will ensure the accumulators and RWST will continue to perform their intended function in the event of an accident.

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

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

Response: No.

The proposed changes include revising accumulator volume and boron concentration requirements and RWST boron concentration requirements that are necessary to accommodate expected changes in the nuclear fuel (*e.g.*, higher enrichment) that are associated with the planned power uprate. Additionally, the change would allow Ginna to utilize analysis methodologies that have been previously approved for use at Westinghouse nuclear plants. The changes to the TS are necessary to ensure the acceptability of these systems to perform their intended function in the event of an accident.

The level of safety of facility operation is not significantly affected by the proposed changes since there is no change in the intent of the TS requirements of assuring proper plant response in the event of an accident. The response of the plant systems to accidents and transients reported in the Updated Final Safety Analysis Report (UFSAR) is not adversely affected by this change. Therefore, the capability to satisfy accident analysis acceptance criteria is not adversely affected. The proposed TS change cannot involve a significant reduction in [a] margin of safety since it is based upon changes that will maintain a substantial margin of safety with respect to accumulators and RWST functions. Therefore, the 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: Daniel F. Stenger, Ballard Spahr Andrews & Ingersoll, LLP, 601 13th Street, NW., Suite 1000 South, Washington, DC 20005.

NRC Section Chief: Richard J. Laufer.

R.E. Ginna Nuclear Power Plant, LLC, Docket No. 50–244, R.E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: April 29, 2005.

Description of amendment request: The amendment would revise Technical Specifications (TSs) to allow the use of Relaxed Axial Offset Control (RAOC) methodology in reducing operator action required to maintain conformance with power distribution control TS and increasing the ability to return to power after a plant trip or transient while still maintaining margin to safety limits under all operating conditions.

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:

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

Response: No.

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

The proposed changes do not initiate an accident. Evaluations and analyses of accidents, which are potentially affected by the parameters and assumptions, associated with the RAOC and $F_Q(Z)$ methodologies have shown that design standards and applicable safety criteria will continue to be met. The consideration of these changes does not result in a situation where the design, material, or construction standards that were applicable prior to the change are altered. Therefore, the proposed changes will not result in any additional challenges to plant equipment that could increase the probability of any previously evaluated accident.

The proposed changes associated with the RAOC and $F_Q(Z)$ methodologies do not affect plant systems such that their function in the control of radiological consequences is adversely affected. The actual plant configurations, performance of systems, or initiating event mechanisms are not being changed as a result of the proposed changes. The design standards and applicable safety criteria limits will continue to be met; therefore, fission barrier integrity is not challenged. The proposed changes associated with the RAOC and $F_Q(Z)$ methodologies have been shown not to adversely affect the plant response to postulated accident scenarios. The proposed changes will therefore not affect the mitigation of the radiological consequences of any accident described in the Updated Final Safety Analysis Report (UFSAR).

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

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

Response: No.

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

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The proposed changes do not challenge the performance or integrity of any safety-related system. The possibility for a new or different type of accident from any accident previously evaluated is not created since the proposed changes do not result in a change to the design basis of any plant structure, system or component. Evaluation of the effects of the proposed changes has shown that design standards and applicable safety criteria continue to be met.

Equipment important to safety will continue to operate as designed and component integrity will not be challenged. The proposed changes do not result in any event previously deemed incredible being made credible. The proposed changes will not result in conditions that are more adverse and will not result in any increase in the challenges to safety systems.

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

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

Response: No.

The proposed changes will not involve a significant reduction in a margin of safety.

The proposed changes will assure continued compliance within the acceptance limits previously reviewed and approved by the NRC for RAOC and $F_Q(Z)$ methodologies. The appropriate acceptance criteria for the various analyses and evaluations will continue to be met.

The projected impact associated with the implementation of RAOC on peak cladding temperature (PCT) has been incorporated into the LOCA [loss-of-coolant accident] analyses for the planned extended power uprate. It has [been] determined that implementation of RAOC at the extended power uprate power level does not result in a significant reduction in a margin of safety. The analysis performed for EPU [extended power uprate] bounds operation at the current power level.

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

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

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

NRC Section Chief: Richard J. Laufer.

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

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the **Federal Register** on the day and page cited. This notice does not extend the notice period of the original notice.

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

Date of amendment request: May 5, 2005.

Brief description of amendment request: The proposed amendment would change the Technical Specifications to modify the auxiliary feedwater (AFW) pump suction protection requirements and change the design basis as described in the Updated Safety Analysis Report to revise the functionality of the discharge pressure switches to provide pump runout protection, which requires operator actions to restore the AFW pumps for specific post-accident recovery activities.

Date of publication of individual notice in ***Federal Register****:* May 13, 2005 (70 FR 25619).

Expiration date of individual notice: June 13, 2005.

PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2 (SSES 1 and 2), Luzerne County, Pennsylvania

Date of amendment request: April 27, 2005, as supplemented May 4, 2005.

Description of amendment request: The proposed amendment would revise the SSES 1 and 2, Technical Specification 3.8.4, ''DC Sources-Operating,'' to address new required actions for the condition in which a 125 volt direct current (VDC) charger is taken out of service for the purposes of a special inspection and related activities. The proposed changes would be in effect until the special inspection and related activities are completed on each of the 125 VDC Class 1E battery chargers but no later than 60 days following the issuance of the Unit 1 and 2 amendments. Specifically, required Action A.2.1 would require that surveillance requirement 3.8.6.1 be performed within 2 hours and once-per-12 hours thereafter; and, required Action A.2.2 would restrict the restoration time for the inoperable electrical power subsystem to 36 hours.

Date of publication of individual notice in ***Federal Register****:* May 12, 2005 (70 FR 25122).

Expiration date of individual notice: Comments, May 27, 2005; Hearing, July 11, 2005.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: October 21, 2004, as supplemented January 4, 2005.

Brief description of amendment: The amendment deleted the Technical Specification (TS) requirements to submit monthly operating reports and annual occupational radiation exposure reports. The change is consistent with Revision 1 of NRC-approved Industry/Technical Specifications Task Force (TSTF) Standard TS Change Traveler, TSTF–369, ''Removal of Monthly Operating Report and Occupational Radiation Exposure Report.'' This TS improvement was announced in the **Federal Register** (69 FR 35067) on June 23, 2004, as part of the Consolidated Line Item Improvement Process (CLIIP).

Date of issuance: May 20, 2005.

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

Amendment No.: 165.

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

Date of initial notice in ***Federal Register***: April 12, 2005 (70 FR 19114).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 20, 2005.

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: February 14, 2005.

Brief description of amendments: The amendments revised the Technical Specification Surveillance Requirement 3.3.7.1 to extend the frequency of the channel functional test for the Engineered Safeguards Protective System digital actuation logic channels from once every 31 days to once every 92 days.

Date of Issuance: May 19, 2005.

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

Amendment Nos.: 345, 347 and 346.

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

Date of initial notice in **Federal Register**: March 15, 2005 (70 FR 12745).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 19, 2005.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 20, 2004, as supplemented by letter dated April 12, 2005.

Brief description of amendment: The amendment deletes TS 6.6.1, ''Occupational Radiation Exposure Report'' and TS 6.6.4, ''Monthly Operating Reports,'' as described in the Notice of Availability published in the **Federal Register** on June 23, 2004 (69 FR 35067).

Date of issuance: May 13, 2005.

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

Amendment No.: 259.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: January 18, 2005 (70 FR 2890). The supplement dated April 12, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 13, 2005.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: December 22, 2004.

Brief description of amendment: The requested change deletes Technical Specification (TS) 6.9.1.5, ''Occupational Radiation Exposure Report,'' and 6.9.1.6, ''Monthly Operating Reports,'' as described in the Notice of Availability published in the **Federal Register** on June 23, 2004 (69 FR 35067).

Date of issuance: May 25, 2005.

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

Amendment No.: 202.

Facility Operating License No. NPF–38: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: March 15, 2005 (70 FR 12746).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 25, 2005.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3 (Waterford 3), St. Charles Parish, Louisiana

Date of amendment request: April 27, 2005, as supplemented by letter dated May 12, 2005.

Brief description of amendment: The amendment removed the license condition on instrument uncertainty that was imposed on the Waterford 3 license with the issuance of License Amendment 199 for the extended power uprate.

Date of issuance: May 23, 2005.

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

Amendment No.: 201.

Facility Operating License No. NPF–38: The amendment revised the Operating License.

Date of initial notice in ***Federal Register***: May 5, 2005 (70 FR 23892). The May 12, 2005, supplemental letter provided clarifying information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 23, 2005.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois; Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: September 15, 2004.

Brief description of amendments: The amendments deleted the Technical Specification (TS) requirements related to hydrogen recombiners. The TS changes support implementation of the revisions to Title 10 of the Code of Federal Regulations (10 CFR) section 50.44, ''Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors,'' that became effective on October 16, 2003. The changes are consistent with Revision 1 of the NRC-approved Industry/Technical Specifications Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–447, ''Elimination of Hydrogen Recombiners and Change to Hydrogen and Oxygen Monitors.''

Date of issuance: May 19, 2005.

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

Amendment Nos.: 137, 137, 143, 143.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 1, 2005 (70 FR 5243).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 19, 2005.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3,York and Lancaster Counties, Pennsylvania

Date of application for amendments: June 15, 2004, as supplemented January 12, 2005.

Brief description of amendments: These amendments changed Surveillance Requirement (SR) 3.8.1.3, monthly diesel surveillance test; SR 3.8.1.10, diesel full load rejection test; SR 3.8.1.14.3.b, diesel 24-hour run test; and, SR 3.8.1.15, diesel hot restart test, to permit these tests to be run at a higher load up to 2800 kW.

Date of issuance: May 20, 2005.

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

Amendments Nos.: 253 and 256.

Renewed Facility Operating License Nos. DPR–44 and DPR–56: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** July 20, 2004, (69 FR 43461). The January 12, 2005, supplement provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register** on July 20, 2004 (69 FR 43461).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 20, 2005.

No significant hazards consideration comments received: No.

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

Date of amendment request: May 21, 2004, as supplemented by letters dated September 16, and December 14, 2004.

Brief description of amendment: The amendment revised the Technical Specification Bases Section to allow the containment spray pumps to be secured during a loss-of-coolant accident, when certain conditions are met, to minimize the potential for containment sump clogging.

Date of issuance: May 20, 2005.

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

Amendment No.: 235.

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

Date of initial notice in ***Federal Register:*** June 22, 2004 (69 FR 34703). The September 16, and December 14, 2004, supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a safety evaluation dated May 20, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 21, 2004.

Brief description of amendment: The amendment revises Technical Specifications related to the reactor coolant pump flywheel inspection program by relocating the requirements from the limiting conditions for operation to the administrative controls section and increasing the inspection interval to 20 years.

Date of issuance: May 9, 2005.

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

Amendment No.: 172.

Renewed Facility Operating License No. NPF–12: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register***: March 1, 2005 (70 FR 9995).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 9, 2005.

No significant hazards consideration comments received: No.

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

Date of amendment request: October 21, 2004, as supplemented December 13 and 22, 2004, and February 23 and March 1, 2005.

Brief description of amendments: Conforming license amendments to remove AEP Texas Central Company as an ''Owner'' in the facility operating licenses.

Date of issuance: May 19, 2005.

Effective date: As of the date of issuance and shall be implemented within 365 days of issuance.

Amendment Nos.: Unit 1–172; Unit 2–160

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

Date of initial notice in ***Federal Register***: December 14, 2004 (69 FR 76019). The supplements dated December 13 and 22, 2004, and February 23 and March 1, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 19, 2005.

No significant hazards consideration comments received: No.

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

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

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

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

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

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

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

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

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

System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR Reference staff at 1 (800) 397–4209, (301) 415–4737 or by e-mail to *pdr@nrc.gov.*

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

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

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

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

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

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

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

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

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

[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.

Tennessee Valley Authority, Docket No. 50–260, Browns Ferry Nuclear Plant, Unit 2, Limestone County, Alabama

Date of amendment request: April 26, 2005, as supplemented on April 29 and on May 3, 2005.

Description of amendment request: Revises the Completion Time for the Action associated with an inoperable low pressure Emergency Core Cooling System injection/spray system to 14 days on a one-time basis.

Date of issuance: May 9, 2005.

Effective date: As of date of issuance and shall be implemented within 7 days.

Amendment No.: 294.

Facility Operating License No. DPR–52: 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, and final determination of NSHC determination are contained in a Safety Evaluation dated May 9, 2005.

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

NRC Section Chief: Michael L. Marshall, Jr.

Dated in Rockville, Maryland, this 27th day of May 2005.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. E5–2848 Filed 6–6–05; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

[Release No. 35–27978]

Notice of Proposal To Amend Articles of Incorporation; Order Authorizing the Solicitation of Proxies

June 1, 2005.

Notice is hereby given that the following filing has been made with the Commission pursuant to provisions of the Act and rules promulgated under the Act. All interested persons are referred to the declaration for complete statements of the proposed transactions summarized below. The declaration and any amendments are available for public inspection through the Commission's Branch of Public Reference.

Interested persons wishing to comment or request a hearing on the declaration should submit their views in writing by June 24, 2005 to the Secretary, Securities and Exchange Commission, Washington DC 20549–0609 and serve a copy on the declarant at the address specified below. Proof of service (by affidavit or, in case of an attorney at law, by certificate) should be filed with the request. Any request for hearing should specifically identify the issues of facts or law that are disputed. A person who so desires will be notified of any hearing, if ordered, and will receive a copy of any notice or order issued in this matter. After June 24, 2005, the declaration, as filed or amended, may be granted or permitted to become effective.

Exelon Corporation (70–10291)

Exelon Corporation (''Exelon''), 10 South Dearborn Street, 37th Floor, Chicago, Illinois, 60603, a registered holding company, has filed a declaration, as amended (''Declaration'') under sections 6(a), 7 and 12(e) of the Public Utility Holding Company Act of 1935 as amended (''Act''), and rules 54 and 62 under the Act.

Exelon seeks authority to amend its Amended and Restated Articles of Incorporation to increase the amount of the Exelon's authorized capital stock and authority to solicit the proxies of the holders of common stock of Exelon.

On December 20, 2004, Exelon and Public Service Enterprise Group Incorporated (''PSEG''), an electric and gas utility holding company that claims exemption from registration pursuant to rule 2 under section 3(a)(1) of the Act, entered into an Agreement and Plan of Merger (''Merger Agreement'').[1] Under the terms of the Merger Agreement, PSEG would merge into Exelon (''Merger''), thereby ending the separate corporate existence of PSEG. Each PSEG shareholder will be entitled to receive 1.225 shares of Exelon common stock for each PSEG share held and cash in lieu of any fraction of an Exelon share that a PSEG shareholder would have otherwise been entitled to receive. Exelon common stock will be unaffected by the Merger, with each issued and outstanding share remaining outstanding following the Merger as a share in the surviving company. Upon completion of the Merger, Exelon will change its name to Exelon Electric & Gas Corporation (''Exelon'').

As the surviving company in the Merger, Exelon will remain the ultimate corporate parent of Commonwealth Edison Company (''ComEd''), PECO Energy Company (''PECO''), Exelon Generation Company, LLC (''Exelon Generation'') and the other Exelon subsidiaries, and become the ultimate corporate parent of Public Service Electric and Gas Company (''PSE&G''), a public utility company under the Act, and the other PSEG subsidiaries.

Exelon will continue to be a registered public utility holding company under the Act, and ComEd, PECO and PSE&G will continue to be operating franchised public utility companies. Exelon will remain headquartered in Chicago, but will also have energy trading and nuclear headquarters in southeastern Pennsylvania and generation headquarters in Newark, New Jersey. PSE&G will remain headquartered in Newark. PECO will remain headquartered in Philadelphia and ComEd will remain headquartered in Chicago.

Under the terms of the Merger Agreement, Exelon and PSEG have agreed to convene meetings of their respective shareholders for the purpose of obtaining required stockholder approvals relating to the Merger. Exelon will seek to obtain the affirmative vote of a majority of votes cast by holders of the outstanding shares of the common stock of Exelon (''Exelon Shares'') represented at the Exelon shareholders meeting (''Exelon Shareholders Meeting'') (provided that at least a majority of the Exelon Shares are represented in person or by proxy at such meeting). Exelon is seeking authority to solicit proxies with respect to proposals for Exelon shareholders to approve the issuance of shares of Exelon common stock as contemplated by the Merger Agreement, and an amendment to Exelon's Amended and Restated Articles of Incorporation to increase the number of authorized shares of Exelon common stock from 1,200,000,000 to 2,000,000,000. In addition, Exelon's shareholders will be asked to vote on the election of five directors to Exelon's Board of Directors, the ratification of the Company's independent accountants for 2005, and the approval of the Exelon 2006 Long-Term Incentive Plan and the Exelon Employee Stock Purchase Plan for Unincorporated Subsidiaries.

Exelon further asks the Commission to issue an order authorizing Exelon to amend its Amended and Restated Articles of Incorporation to increase the number of authorized shares of Exelon common stock from 1,200,000,000 to 2,000,000,000.

Fees and expenses in the estimated amount of $2,140,750.00 are expected by Exelon to be incurred in connection

[1] The Merger is subject to a number of conditions, including the approval of the Commission under the Act and other regulatory approvals. On March 15, 2005 Exelon filed an application with this Commission seeking approval of the Merger and related transactions. SEC File No. 70–10294.