Program Directors (CRCPD) (Public Meeting) (Contact: Shawn Smith, 301–415–2620).

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

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

Week of September 18, 2006—Tentative

There are no meetings scheduled for the Week of September 18, 2006.

* 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: Michelle Schroll, (301) 415–1662.

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

The NRC provides reasonable accommodation to individuals with disabiloities where appropriate. If you need a reasonable acommodation 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, Deborah Chan, at 301–415–7041, TDD: 301–415–2100, or by E-mail at *DLC@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: August 10, 2008.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 06–6939 Filed 8–11–06; 9:59 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from July 21, 2006, to August 3, 2006. The last biweekly notice was published on August 1, 2006 (71 FR 43528).

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.*

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

Date of amendment request: June 30, 2006.

Description of amendment request: The proposed change would revise the Note preceding Technical Specification (TS) Surveillance Requirement (SR) 3.4.6.1 to be consistent with the wording in NUREG–1434, ''Standard Technical Specifications General Electric Plants, BWR/6,'' Revision 3.

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

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

Response: No.

The proposed amendment revises the note associated with TS SR 3.4.6.1, which requires verification that the leakage past the Reactor Coolant System (RCS) Pressure Isolation Valves (PIVs) is less than a specified limit. The proposed revision provides clarification that performance of this SR is allowed during plant shutdown (*i.e.*, a Mode other than Modes 1 and 2).

The proposed change does not require modification to the facility. The proposed change does not affect the operation of any facility equipment, the interface between facility systems, or the reliability of any equipment. In addition, the proposed change does not alter the requirement to perform the leakage testing of the RCS PIVs and does not revise the leakage limits associated with this SR. The function of the RCS PIVs is to separate the high pressure RCS from an attached low pressure system. Periodic testing of PIVs can substantially reduce intersystem Loss of Coolant Accident (LOCA) probability. Since the proposed change does not alter the method or limits associated with the leak rate testing of the RCS PIVs there is no significant increase in the probability of a LOCA. Therefore, the proposed amendment does not involve a significant increase in the probability of an accident previously evaluated.

The consequences of a previously analyzed event are dependent on the initial conditions assumed in the analysis, the availability and successful functioning of equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. The method for performing the leakage testing of the RCS PIVs and the specified leakage limit for this testing will not change as a result of the proposed revision and, therefore, there is no change in the consequences associated with the LOCA analysis. The radiological consequences remain within applicable regulatory limits. The proposed change does not alter any system's performance measures or the ability to perform its accident mitigation functions. The radiological consequences associated with any previously evaluated accident do not change as a result of the proposed revision. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

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

Response: No.

The proposed change to the wording of the Note to TS SR 3.4.6.1 clarifies the plant conditions for when the surveillance is required to be performed. The proposed change does not affect the design, functional performance or operation of the facility. No new equipment is being introduced and installed equipment is not being operated in a new or different manner. Similarly, the proposed change does not affect the design or operation of any structures, systems or components involved in the mitigation of any accidents, nor does it affect the design or operation of any component in the facility such that new equipment failure modes are created. There are no setpoints at which protective or mitigative actions are initiated that are affected by this proposed action. No change is being made to procedures relied upon to respond to an off-normal event.

As such the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

Margins of safety are established in the design of components, the configuration of components to meet certain performance parameters, and in the establishment of setpoints to initiate alarms or actions. The proposed change revises a note associated with a surveillance requirement to clarify the plant conditions for when the surveillance needs to be performed. This change involves an administrative clarification to reflect the original intent of the TS. The equipment will continue to be tested in a manner and at a frequency necessary to provide confidence that the equipment can perform its intended safety function. There is no change in the design of the affected systems, no alteration of the setpoints at which alarms or actions are initiated, and no change in plant configuration from original design. There is no impact on the plant safety analyses.

Therefore, operation of CPS in accordance with the proposed change will 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. Bradley J. Fewell, Assistant General Counsel, Exelon Generation Company, LLC, 200 Exelon Way, Kennett Square, PA 19348.

NRC Branch Chief: Daniel S. Collins.

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

Date of amendment request: June 14, 2006.

Description of amendment request: The proposed change will delete Waterford 3 Technical Specification (TS) Surveillance Requirement (SR) 4.8.1.1.2.f. This SR requires that the emergency diesel generator be subjected to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

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 ability of the emergency diesel generator to perform its safety function is not proven by the performance of the manufacturer's recommended inspections. The inspections are not considered an initiator or mitigating factor in any previously evaluated accidents.

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

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

Response: No.

The proposed change results in the deletion of the SR associated with the performance of manufacturer's inspections. No modifications to plant structures, systems, or components, or changes in the design of the plant structures, systems, or components are required to support the proposed TS change.

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 ability of the emergency diesel generator to perform its safety function is not proven by the performance of the manufacturer's recommended inspections. Inspection activities will continue to be performed.

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: N.S. Reynolds, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: David Terao.

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

Date of amendment request: June 2, 2006.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) Surveillance Requirement (SR) 3.4.3.1 to increase the allowable as-found main steam safety valve (MSSV) lift set point tolerance from +/−1 percent to +/−3 percent. The proposed change would also revise the SR 3.1.7.10 to increase the enrichment of sodium pentaborate used in the Standby Liquid Control (SLC) system from greater than or equal to 30 atom percent boron-10 to greater than or equal to 45 atom percent boron-10.

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 increases the allowable as-found MSSV lift setpoint tolerance, determined by test after the valves have been removed from service, from +/−1 percent to +/−3 percent. The proposed change does not alter the TS requirements for the number of MSSVs required to be operable, the nominal lift setpoints, the allowable as-left lift setpoint tolerance, the MSSV testing frequency, or the manner in which the valves are operated.

Consistent with current TS requirements, the proposed change continues to require that the MSSVs be adjusted to within +/−1 percent of their nominal lift setpoints following testing. Since the proposed change does not alter the manner in which the valves are operated, there is no significant impact on reactor operation.

The proposed change does not involve a physical change to the valves, nor does it change the safety function of the valves. The proposed TS revision involves no significant changes to the operation of any systems or components in normal or accident operating conditions and no changes to existing structures, systems, or components, with the exception of the SLC system enrichment change. The proposed change to increase the enrichment of sodium pentaborate used in the SLC system by a design modification using a single SLC pump will ensure that the requirements of 10 CFR 50.62, ''Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants,'' continue to be met. The SLC system is not an initiator to an accident; rather, the SLC system is used to mitigate a postulated anticipated transient without scram (ATWS) event. Therefore, these changes will not increase the probability of an accident previously evaluated.

Generic considerations related to the change in setpoint tolerance were addressed in NEDC–31753P, ''BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report,'' and were reviewed and approved by the NRC in a safety evaluation dated March 8, 1993. The plant specific evaluations, required by the NRC's safety evaluation and performed to support this proposed change, show that there is no change to the design core thermal limits and adequate margin to the reactor vessel pressure limits using a +/−3 percent lift setpoint tolerance. These analyses also show that operation of Emergency Core Cooling Systems is not affected, and the containment response following a loss-of-coolant accident is acceptable. The plant systems associated with these proposed changes are capable of meeting applicable design basis requirements and retain the capability to mitigate the consequences of accidents described in the Updated Final Safety Analysis Report. Therefore, these changes do not involve an increase in the consequences of an accident previously evaluated.

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 increases the allowable as-found lift setpoint tolerance for the DNPS MSSVs, and increases the required enrichment of sodium pentaborate used in the SLC system. The proposed change to increase the enrichment of sodium pentaborate used in the SLC system will ensure that the requirements of 10 CFR 50.62 continue to be met.

The proposed change to increase the MSSV tolerance was developed in accordance with the provisions contained in the NRC safety evaluation for NEDC–31753P. MSSVs installed in the plant following testing or refurbishment will continue to meet the current tolerance as-left acceptance criteria of +/−1 percent of the nominal setpoint. The proposed change does not affect the manner in which the overpressure protection system is operated; therefore, there are no new failure mechanisms for the overpressure protection system.

The proposed change to allow an increase in the MSSV setpoint tolerance does not alter the nominal MSSV lift setpoints or the number of MSSVs currently required to be operable by DNPS TS. The proposed change does not involve physical changes to the valves, nor does it change the safety function of the valves. There is no alteration to the parameters within which the plant is normally operated. As a result, no new failure modes are being 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?

Response: No.

The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The proposed change does not modify the safety limits or setpoints at which protective actions are initiated, and does not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety.

Establishment of the ±3 percent MSSV setpoint tolerance limit does not adversely impact the operation of any safety-related component or equipment. Evaluations performed in accordance with the NRC safety evaluation for NEDC–31753P have concluded that all design limits will continue to be met.

The proposed change to increase the enrichment of sodium pentaborate used in the SLC system will ensure that the requirements of 10 CFR 50.62 continue to be met.

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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Mr. Bradley J. Fewell, Assistant General Counsel, Exelon Generation Company, LLC, 200 Exelon Way, Kennett Square, PA 19348.

NRC Branch Chief: Daniel S. Collins.

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

Date of amendment request: March 16, 2006.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) 3.3.6.1, ''Primary Containment Isolation Instrumentation,'' Table 3.3.6.1–1 to revise the allowable values (AVs) for the reactor core isolation cooling (RCIC) temperature-based leak detection. The proposed change is a result of revising the setpoint calculation for the subject temperature instruments based on the current reactor coolant leak detection analytical limit. The temperature limits correspond to a 25-gallon per minute (gpm) leak as determined by LSCS calculations. The proposed changes would revise TS Table 3.3.6.1–1 AVs for the following four RCIC system isolation functions:

Item 3.e. RCIC Equipment Room Temperature—High
Item 3.f. RCIC Equipment Room Differential Temperature—High
Item 3.g. RCIC Steam Line Tunnel Temperature—High
Item 3.h. RCIC Steam Line Tunnel Differential Temperature—High

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 is a result of revising the setpoint calculation for the subject temperature instruments based on the current reactor coolant leak detection calculation analytical limit. The proposed changes will revise TS Table 3.3.6.1–1 Allowable Values for the following four RCIC system isolation functions as noted below.

- Increase the Allowable Value for Function 3.e., ''RCIC Equipment Room Temperature—High,'' from ≤ 291.0 °F to ≤ 297.0 °F
- Decrease the Allowable Value for Function 3.f., ''RCIC Equipment Room Differential Temperature—High,'' from ≤ 189.0 °F to ≤ 188.0 °F
- Decrease the Allowable Value for Function 3.g., ''RCIC Steam Line Tunnel Temperature—High,'' from ≤ 277.0 °F to ≤ 267.0 °F
- Increase the Allowable Value for Function 3.h., ''RCIC Steam Line Tunnel Differential Temperature—High,'' from ≤ 155.0 °F to ≤ 163.0 °F

The function of the instrumentation listed on TS Table 3.3.6.1–1, in combination with other accident mitigation features, is to limit fission product release during and following postulated Design Basis Accidents to within allowable limits. The Allowable Values specified in TS Table 3.3.6.1–1 provide assurance that the instrumentation will perform as designed.

The Allowable Values for RCIC system isolation are not a precursor to any accident previously evaluated. Accidents are assumed to be initiated by equipment failure. The proposed change does not alter the initiation conditions or operational parameters for the system. There is no increase in the failure probability of the system. As such, the probability of occurrence for a previously evaluated accident is not increased.

The Allowable Values specified in Table 3.3.6.1–1 provide assurance that the RCIC system will perform as designed. The proposed revision to the Allowable Values does not change any of the RCIC system leak detection isolation actuation setpoints. Thus, the radiological consequences of any accident previously evaluated are not increased.

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

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

The proposed change does not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The proposed change does not change or introduce any new equipment, modes of system operation or failure mechanisms.

The proposed change is based on revised reactor coolant leak detection calculation analytical limits determined by the most current revision to the heat rise calculation. Setpoint calculations have been performed to determine the nominal trip setpoints and Allowable Values for the instrumentation associated with the leak detection function based on the revised analytical limits determined by the heat rise calculations. The proposed revision to the Allowable Values does not change any of the RCIC system leak detection isolation actuation setpoints.

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

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

The proposed change will revise TS Table 3.3.6.1–1 Allowable Values for the instrument functions associated with RCIC Isolation.

The current Allowable Values for these functions are:

≤ 291.0 °F for RCIC Equipment Room Temperature—High
≤ 189.0 °F for RCIC Equipment Room Differential Temperature—High
≤ 277.0 °F for the RCIC Steam Line Tunnel Temperature—High
≤ 155.0 °F for the RCIC Steam Line Tunnel Differential Temperature—High

The proposed change revises the Allowable Values to the following:

≤ 297.0 °F for RCIC Equipment Room Temperature—High
≤ 188.0 °F for RCIC Equipment Room Differential Temperature—High
≤ 267.0 °F for the RCIC Steam Line Tunnel Temperature—High
≤ 163.0 °F for the RCIC Steam Line Tunnel Differential Temperature—High

The proposed change is a result of revising the setpoint calculation for the subject temperature instruments based on the current analytical limit. The proposed changes will revise TS Table 3.3.6.1–1 Allowable Values for the subject four RCIC system isolation functions and will provide assurance that the RCIC system will perform as designed. The proposed revision to the Allowable Values does not change any of the RCIC system leak detection isolation actuation setpoints.

Margin of safety is established by the design and qualification of plant equipment, the operation of the plant within analyzed limits, and the point at which protective or mitigative actions are being initiated. The proposed change does not alter these considerations. The proposed allowable values will still ensure that the results of the accident analysis remain valid.

Based on this information, 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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Mr. Bradley J. Fewell, Assistant General Counsel, Exelon Generation Company, LLC, 200 Exelon Way, Kennett Square, PA 19348.

NRC Branch Chief: Daniel S. Collins.

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 4, 2006.

Description of amendment request: The proposed amendment request will add one NRC approved topical report reference to the list of analytical methods in Technical Specification (TS) 5.6.5, ''Core Operating Limits Report (COLR),'' that can be used to determine core operating limits, and will delete seven obsolete references from the same TS Section.

The proposed changes are:

1. Add an NRC previously approved Topical Report ANF–1358(P)(A), Revision 3, ''The Loss of Feedwater Heating Transient in Boiling Water Reactors,'' (LOFWH), which will list FRA–ANP method for evaluating the LOFWH transient.

2. Delete seven references describing previously approved Global Nuclear Fuel (GNF) and FRA–ANP methodologies for the analyses of ATRIUM–9B and GE9 fuel. Both of these fuel types have been or will be completely discharged from both Lasalle County Station (LSCS) reactors after the loading of ATRIUM–10 fuel during the LSCS Unit 2 refuel outage currently scheduled to begin in February 2007 (*i.e.*, L2R11).

The proposed changes support the continued irradiation of ATRIUM–10 fuel in the LSCS reactors and the use of the NRC-approved analytical methodology for evaluation of LOFWH transients.

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

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

Response: No.

Technical Specification (TS) 5.6.5 lists NRC-approved analytical methods used at LaSalle County Station (LSCS) to determine core operating limits. The proposed changes will add an NRC-approved topical report reference to the list of administratively controlled analytical methods in TS 5.6.5, ''Core Operating Limits Report (COLR),'' that can be used to determine core operating limits, and delete seven obsolete references.

The addition of a Framatome ANP (FRA–ANP) methodology to determine overall core operating limits for future LSCS core configurations was approved by the NRC in Reference 2. LSCS Unit 2 will continue to load Framatome ANP ATRIUM–10 fuel during the Unit 2 Refueling Outage 11 currently scheduled for February 2007. The proposed change to TS 5.6.5 will add a FRA–ANP methodology as a reference to determine core operating limits for loss of feedwater heater (LOFWH) conditions. Thus, the proposed change will allow LSCS to use the most recent FRA–ANP methodology for analysis of LOFWH conditions.

The addition and deletion of approved analytical methods in TS Section 5.6.5 has no effect on any accident initiator or precursor previously evaluated and does not change the manner in which the core is operated. The NRC-approved methods ensure that the output accurately models predicted core behavior, have no effect on the type or amount of radiation released, and have no effect on predicted offsite doses in the event of an accident. Additionally, the NRC-approved methods do not change any key core parameters that influence any accident consequences. Thus, the proposed changes do not have any effect on the probability of an accident previously evaluated.

The methodology conservatively establishes acceptable core operating limits such that the consequences of previously analyzed events are not significantly increased.

The proposed changes in the list of analytical methods do not affect the ability of LSCS to successfully respond to previously evaluated accidents and does not affect radiological assumptions used in the evaluations. Thus, the radiological consequences of any accident previously evaluated are not increased.

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

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

Response: No.

The proposed changes to TS Section 5.6.5 do not affect the performance of any LSCS structure, system, or component credited with mitigating any accident previously evaluated. The NRC-approved analytical methodology for evaluating LOFWH transients will not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The proposed changes do not introduce any new modes of system operation or failure mechanism.

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

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

Response: No.

The proposed changes will add a reference to the list of analytical methods in TS 5.6.5 that can be used to determine core operating limits and delete seven obsolete references. The proposed changes do not modify the safety limits or setpoints at which protective actions are initiated and do not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety. Therefore, the proposed changes provide an equivalent level of protection as that currently provided.

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

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

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

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 8, 2006.

Description of amendment request: The proposed changes modify Technical Specifications (TSs) 3.1.3, ''Control Rod OPERABILITY''; 3.1.6, ''Rod Pattern Control''; 3.3.2.1, ''Control Rod Block Instrumentation''; 3.10.7, ''Control Rod Testing—Operating''; and 3.10.8, ''SHUTDOWN MARGIN (SDM) Test—Refueling'' to replace the current references to banked position withdrawal sequence (BPWS) with references to ''the analyzed rod position sequence.''

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

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

Response: No.

The proposed change modifies Technical Specifications (TS) 3.1.3, ''Control Rod OPERABILITY''; TS 3.1.6, ''Rod Pattern Control''; TS 3.3.2.1, ''Control Rod Block Instrumentation''; TS 3.10.7, ''Control Rod Testing—Operating'', and; TS 3.10.8, SHUTDOWN MARGIN (SDM) Test—Refueling''. The proposed change would replace the current references to ''Banked Position Withdrawal Sequence (BPWS)'' with references to ''the analyzed rod position sequence''. The use of the ''the analyzed rod position sequence'' will continue to minimize the consequences of an accident previously evaluated including the Control Rod Drop Accident (CRDA). Additionally, the use of the words ''the analyzed rod position sequence'' will provide an equivalent level of protection during plant startups and shutdowns and therefore will not increase the consequences of an accident previously evaluated.

Control rod patterns during startup and shutdown conditions will continue to be controlled by the operator and the Rod Worth Minimizer (RWM) (LCO [limiting condition of operation] 3.3.2.1, ''Control Rod Block Instrumentation''), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10% of Rated Thermal Power. As a result of this change, these sequences will continue to limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA).

Accidents are initiated by the malfunction of plant equipment, or the failure of plant structures, systems, or components. The proposed change will ensure that analyzed rod position sequences are developed to minimize incremental control rod reactivity worth in accordance with the ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–15 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–15–US, September, 2005, NRC approved methodology, and reviewed and approved in accordance with the 10 CFR 50.59 process. These analyzed rod position sequences will limit the potential reactivity increase for a postulated CRDA during reactor startups and shutdowns below the Low Power Setpoint of 10% of Rated Thermal Power.

The proposed change will continue to ensure that systems, structures and components are capable of performing their intended safety functions.

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

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

Response: No.

The proposed change does not affect the assumed accident performance of the control rods, nor any plant structure, system, or component previously evaluated.

The proposed change does not involve the installation of new equipment, and installed equipment is not being operated in a new or different manner. The change ensures that control rods remain capable of performing their safety functions. No set points are being changed which would alter the dynamic response of plant equipment. Accordingly, no new failure modes are introduced.

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

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

Response: No.

The proposed change will ensure that analyzed rod position sequences are developed to minimize incremental control rod reactivity worth in accordance with the ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–15 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–15–US, September, 2005, NRC approved methodology, and reviewed and approved in accordance with the 10 CFR 50.59 process. The proposed change will not adversely impact the plant's response to an accident or transient. All current safety margins will be maintained. There are no changes proposed which alter the set points at which protective actions are initiated, and there is no change to the operability requirements for equipment assumed to operate for accident mitigation.

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. Brad Fewell, Assistant General Counsel, Exelon Generation Company, LLC, 200 Exelon Way, Kennett Square, PA 19348.

NRC Branch Chief (Acting): Brooke D. Poole.

FirstEnergy Nuclear Operating Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS–1 and 2), Beaver County, Pennsylvania

Date of amendment request: June 14, 2006.

Description of amendment request: The amendments would incorporate the results of a new spent fuel pool criticality analysis documented in WCAP–16518–P/WCAP–16518–NP, ''Beaver Valley Unit 2 Spent Fuel Pool Criticality Analysis,'' Revision 1, May 2006 for the BVPS–2 spent fuel storage pool. The revised criticality analysis will permit utilization of vacant storage locations dictated by the existing Technical Specification (TS) storage configurations in the BVPS–2 spent fuel storage pool.

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 relevant accidents previously evaluated are limited to the fuel handling and criticality accidents.

Administrative controls during fuel fabrication ensure that the fuel is fabricated to ensure proper loading of fuel in the fuel assemblies. Administrative and operational controls used to load fuel assemblies into the spent fuel pool ensure the fuel assemblies are stored in compliance with the allowed storage configurations. Fuel handling is performed under administrative controls and physical limitations. These controls will remain in effect and continue to protect against criticality and fuel handling accidents involving new storage configurations dictated by the new analysis. There is therefore no impact on the probability of fuel handling or criticality accidents.

The new criticality analysis defines new spent fuel storage configurations with new enrichment and burnup limits. Integral Fuel Burnable Absorber (IFBA) limits are used to comply with the 1-out-of-4 configuration for fresh fuel. The boron dilution evaluation that supported Amendment [No.] 128 [February 11, 2002, Agencywide Documents Access and Management System Accession No. ML020020373], permitting credit for soluble boron at BVPS Unit No. 2 continues to remain valid. The new analysis demonstrates that k_{eff} remains below unity for the various storage configurations considered with zero soluble boron, and that k_{eff} remains less than or equal to 0.95 for the entire pool with credit for soluble boron under non-accident and accident conditions with a 95% probability at a 95% confidence level (95/95). Potential consequences of accidents previously analyzed remain unchanged.

The editorial changes made to the table numbers and the LCO [Limiting Condition for Operation] and Surveillance Requirement wording do not impact probability or consequences of an accident previously evaluated.

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

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

Response: No. The relevant types of accidents previously evaluated are limited to criticality and fuel handling accidents. Although the new analysis will allow utilization of additional storage capacity, implementation of fuel loading configurations and fuel handling activities will continue to be performed under administrative and operational controls. No new or different activities are introduced as a result of the proposed changes. The utilization of additional storage capacity within the allowances of the revised analysis will introduce no new or other kind of accident.

The editorial changes made to the table numbers and the LCO and Surveillance Requirement wording do not impact any previously evaluated accident.

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

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

Response: No. The margin to safety with respect to analyzed accidents involves maintaining k_{eff} through fuel storage configurations and boron concentration controls in the spent fuel pool. The boron dilution evaluation that supported that supported Amendment [No.] 128 permitting credit for soluble boron at BVPS Unit No. 2 remains valid. The Amendment [No.] 128 evaluation concluded that a boron dilution event is not credible for BVPS Unit No. 2. The new analysis calculates the non-accident soluble boron concentration to be less than was determined in the Amendment [No.] 128 evaluation. Thus, there is no significant reduction in a margin of safety because of the new analysis and the conclusions of the Amendment [No.] 128 dilution evaluation remain valid.

Under accident conditions, the soluble boron needed to maintain k_{eff} below 0.95 with the new storage configurations is less than what is assumed in current analysis. The proposed change does not involve a significant reduction in a margin of safety for accident conditions.

The editorial changes made to the table numbers and the LCO and Surveillance Requirement wording do not impact a margin of safety.

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

NRC Branch Chief: Richard J. Laufer.

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

Date of amendment request: June 1, 2006.

Description of amendment request: The proposed amendment would modify Technical Specification 3.4.10, ''Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown'' by adding a default Condition to address situations when an RHR shutdown cooling subsystem becomes inoperable in MODE 4 and, within the completion time of 1 hour, an alternate method of decay heat removal can not be verified to be available.

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

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

Response: No. The proposed amendment does not change the design of any structures, systems or components (SSCs), and does not affect the manner in which plant systems are operated. It is a change to the Technical Specifications only, to provide guidance to plant operators on appropriate actions to take, where no Technical Specification guidance currently exists. Since the design of plant SSCs is not changed and plant systems and components are not operated in a different manner, there is no change to previously identified accident initiators, and the proposed amendment would not impact the probability of any of the previously evaluated accidents in the Updated Safety Analysis Report (USAR).

The USAR event that evaluates the consequences of a loss of RHR Shutdown Cooling is included in Section 15.2.9 entitled ''Failure of RHR Shutdown Cooling''. This event examines the consequences of a loss of not only an RHR shutdown cooling subsystem, but also the loss of the suction source from the recirculation system leading to both RHR Shutdown Cooling subsystems, and a loss of offsite power. Even with these multiple failures, this event is not one of the limiting transients. As noted in Section 15.2.9.5, ''Radiological Consequences,'' there are no fuel failures, and the consequences of the event are much less than those for the ''Main Steam Isolation Valve Closure'' transient, which is evaluated with acceptable results in USAR Section 15.2.4.5. Since the proposed amendment only involves the addition of a Required Action where no guidance currently exists, and the design of plant SSCs is not changed and plant systems and components are not operated in a different manner, the proposed amendment does not affect the consequences of the Section 15.2.9 analysis, nor does it affect the ability of the installed RHR subsystems to perform their shutdown cooling function. The change adds a default Condition to provide guidance to the operators in those situations when a subsystem becomes inoperable with the plant in MODE 4 and an alternate cannot be verified to be available within an hour, which does not impact the consequences of the previously evaluated accidents in the USAR.

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

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

Response: No. This change to the required Technical Specification actions does not involve a change in the design function or operation of plant SSCs. It does not introduce credible new failure mechanisms, malfunctions, or accident initiators not considered in the existing plant design and licensing basis.

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 amendment involve a significant reduction in a margin of safety?

Response: No. This proposed amendment only involves a change to the required Technical Specification actions. It does not involve a change in the evaluation and analysis methods used to demonstrate compliance with regulatory and licensing requirements, and does not exceed or alter a design basis or safety limit. The safety margin before the change remains unchanged after the proposed amendment.

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 W. Jenkins, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Branch Chief: Daniel S. Collins.

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

Date of amendment request: June 1, 2006.

Description of amendment request: The proposed amendment would modify Technical Specification 3.4.9, ''Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown,'' to revise the Required Actions when both RHR shutdown cooling subsystems are inoperable in MODE 3.

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

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

Response: No. The proposed amendment does not change the design of any structures, systems or components (SSCs), and does not affect the manner in which plant systems are operated. It is a change to the Technical Specifications only, to provide guidance to plant operators on appropriate actions to take, when both RHR shutdown cooling subsystems are inoperable. Since the design of plant SSCs is not changed and plant systems and components are not operated in a different manner, there is no change to previously identified accident initiators, and the proposed amendment would not impact the probability of any of the previously evaluated accidents in the Updated Safety Analysis Report (USAR).

The USAR event that evaluates the consequences of a loss of RHR Shutdown Cooling is included in Section 15.2.9 entitled ''Failure of RHR Shutdown Cooling.'' This event examines the consequences of a loss of not only an RHR shutdown cooling subsystem, but also the loss of the suction source from the recirculation system leading to both RHR Shutdown Cooling subsystems, and a loss of offsite power. Even with these multiple failures, this event is not one of the limiting transients. As noted in Section 15.2.9.5, ''Radiological Consequences,'' there are no fuel failures, and the consequences of the event are much less than those for the ''Main Steam Isolation Valve Closure'' transient, which is evaluated with acceptable results in USAR Section 15.2.4.5. Since the proposed amendment only involves the addition of a Required Action where no guidance currently exists, and the design of plant SSCs is not changed and plant systems and components are not operated in a different manner, the proposed amendment does not affect the consequences of the Section 15.2.9 analysis, nor does it affect the ability of the installed RHR subsystems to perform their shutdown cooling function.

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

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

Response: No. This change to the required Technical Specification actions does not involve a change in the design function or operation of plant SSCs. It does not introduce credible new failure mechanisms, malfunctions, or accident initiators not considered in the existing plant design and licensing basis.

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 amendment involve a significant reduction in a margin of safety?

Response: No. This proposed amendment only involves a change to the required Technical Specification actions. It does not involve a change in the evaluation and analysis methods used to demonstrate compliance with regulatory and licensing requirements, and does not exceed or alter a design basis or safety limit. The safety margin before the change remains unchanged after the proposed amendment.

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 W. Jenkins, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Branch Chief: Daniel S. Collins.

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 28, 2006.

Description of amendment request: The proposed amendment would change the SSES 1 and 2 Technical Specifications (TSs) to modify the standby liquid control system for single loop pump operation and use of enriched sodium pentaborate solution.

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:

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

Response: No.

The proposed changes revise Technical Specification 3.1.7 for the Standby Liquid Control (SLC) system to reflect new boron weight-percent and enrichment requirements. In addition, the change to single pump operation reduces the required SLC pump flow and discharge pressure required to satisfy 10 CFR 50.62, thus increasing the reliability of the system. The changes do not otherwise alter the design or operation of the SLC system, and the existing design of the system is sufficient to support operation with the enriched sodium pentaborate solution. The SLC system is not considered to be the initiator of any event currently analyzed in the FSAR [Final Safety Analysis Report]. Therefore, the proposed changes do not increase the probability of a previously evaluated accident.

The SSES ATWS [anticipated transient without scram] analysis was performed using standard accepted assumptions, inputs, and codes. That analysis, which demonstrated that the acceptance criteria for peak vessel pressure, peak cladding temperature, peak local cladding oxidation, peak suppression pool temperature, and peak containment pressure, established the requirements for the proposed boron weight-percent and concentration, and pump flow rate. The analysis assumed the use of only a single pump, versus two pumps. The results of the analysis are that no fission product barriers are adversely challenged, and the radiological consequences of previously evaluated accidents (i.e., ATWS) are not increased.

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

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

Response: No.

The proposed changes revise Technical Specification 3.1.7 for the SLC system to reflect new boron weight-percent and enrichment requirements. In addition, the change to single pump operation reduces the required SLC pump flow and discharge pressure required to satisfy 10 CFR 50.62, thus increasing the reliability of the system. A new Surveillance Requirement (SR 3.1.7.10) is also added to verify the correct solution enrichment prior to addition of inventory to the SLC tank. The changes do not otherwise alter the design or operation of the SLC system, and the existing design of the system is sufficient to process the enriched sodium pentaborate solution. With the exception of these changes, no other physical changes to plant structures or systems are proposed. Thus, the proposed changes do not create a new initiating event for the spectrum of events currently postulated in the FSAR.

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

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

Response: No.

The proposed changes revise Technical Specification 3.1.7 for the SLC system to reflect new boron weight-percent and enrichment requirements. In addition, the change to single pump operation reduces the required SLC pump flow and discharge pressure required to satisfy 10 CFR 50.62, thus increasing the reliability of the system. The changes do not otherwise alter the design or operation of the SLC system, and the existing design of the system is sufficient to process the enriched sodium pentaborate solution.

The analysis was performed using standard accepted assumptions, inputs, and codes. That analysis, which demonstrated that ATWS acceptance criteria are satisfied, established the requirements for the proposed boron weight-percent and concentration, and pump flow rate. Further, the analysis assumed only a single pump is in operation verses two pumps. The evaluation demonstrated that the SLC system meets this post-LOCA [loss-of-coolant accident] suppression pool pH control design function.

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: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Branch Chief: Richard J. Laufer.

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

Date of amendment request: October 12, 2004.

Description of amendment request: As part of Nuclear Regulatory Commission's (NRC) approval of the Improved Technical Specifications for Browns Ferry Nuclear Plant, Unit 1, by Amendment No. 234, NRC imposed License Condition 2.C(4) to ensure that the required analyses and modifications needed to support the Technical Specification (TS) changes made by License Amendment No. 234 and any subsequent TS changes, were completed by licensee prior to entering the mode for which the TS applies. The proposed amendment would remove this license condition from the license.

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

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

Response: No.

The proposed amendment does not affect any precursors for accidents described in Chapter 14 of the Browns Ferry Updated Final Safety Analysis Report (UFSAR). The proposed amendment does not change the conditions, operating configurations, or minimum amount of operating equipment assumed in the safety analysis for accident mitigation. No changes are proposed in plant protection or which create new modes of plant operation. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed amendment does not introduce new equipment, which could create a new or different kind of accident. No new external threats, release pathways, or equipment failure modes are created. Therefore, the implementation of the proposed amendment will not create a possibility for an accident of a new or different type than those previously evaluated.

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

Response: No.

The proposed amendment does not impact the redundancy or availability of equipment credited in the response to accidents described in Chapter 14 of the UFSAR. For these reasons, the proposed amendment does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: L. Raghavan.

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

Date of amendment request: May 1, 2006 (TS–455).

Description of amendment request: The proposed amendment would revise the numeric values of the safety limit minimum critical power ratio (SLMCPR) in the Technical Specification (TS) Section 2.1.1.2 for single and two reactor recirculation loop operation to incorporate the results of the Browns Ferry Nuclear Plant, Unit 1 Cycle 7 SLMCPR analysis.

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 Technical Specification change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed amendment establishes a revised SLMCPR value for single and two recirculation loop operation. The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The proposed SLMCPR values preserve the existing margin to transition boiling and the probability of fuel damage is not increased. Since the change does not require any physical plant modifications or physically affect any plant components, no individual precursors of an accident are affected and the probability of an evaluated accident is not increased by revising the SLMCPR values.

The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. The revised SLMCPR values have been determined using NRC-approved methods and procedures. The basis of the MCPR Safety Limit is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. These calculations do not change the method of operating the plant and have no effect on the consequences of an evaluated accident. Therefore, the proposed TS change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed license amendment involves a revision of the SLMCPR value for single and two recirculation loop operation based on the results of an analysis of the Unit 1 Cycle 7 core. Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in the allowable methods of operating the facility. This proposed license amendment does not involve any modifications of the plant configuration or changes in the allowable methods of operation. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident previously evaluated.

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

Response: No.

The margin of safety as defined in the TS bases will remain the same. The new SLMCPR values were calculated using referenced fuel vendor methods and procedures, which are in accordance with the fuel design and licensing criteria. The SLMCPR remains high enough to ensure that greater than 99.9 percent of all fuel rods in the core are expected to avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity. Therefore, the proposed TS change does not involve a reduction in the margin of safety.

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

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

NRC (Acting) Branch Chief: L. Raghavan.

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

Date of amendment request: July 6, 2006 (TS–06–04).

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) for the Sequoyah Nuclear Plant, Units 1 and 2. Action a.1 of TS 3.1.3.2, ''Position Indication Systems—Operating,'' requires the verification of rod position by use of the moveable incore detectors. Tennessee Valley Authority (the licensee, TVA) is proposing a revision to TS 3.1.3.2 to allow the position of the control and shutdown rods to be monitored by a means other than the moveable incore detectors. The amendment will provide a less burdensome monitoring method should problems with the analog rod position indication (ARPI) system be experienced. When a recurring problem in the system requires the monitoring of a rod's position by the alternate means, TVA plans to continue unit operation and to use the alternate means until the unit enters Mode 5 and repairs to the system can safely be implemented.

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 provides an alternative method for the monitoring of the position of a rod once the position of the rod is verified using the moveable incore detector system. The proposed monitoring of rod control system parameters provides a reasonably similar approach to rod position monitoring as that provided by the movable incore detector system. In particular, the ability to immediately detect a rod drop or misalignment is not directly provided by the movable incore detector system or by the monitoring of rod control system parameters. Additionally, neither the movable incore detector system, nor the monitoring of rod control system parameters, provides the capability to verify rod position following a reactor trip or shutdown. Therefore, the monitoring of rod control system parameters, in lieu of the use of the movable incore detector system, provides an equivalent and acceptable method of monitoring rod position while a position indicator is inoperable.

The proposed change does not alter plant equipment that is considered to have the potential to alter the probability of an accident. The affected components are for monitoring only and do not actively affect equipment that interacts with the control of the reactor. Likewise, the affected components are for monitoring and provide an equivalent level of indication of rod position as the current action. This maintains an acceptable level of rod position indication for normal plant operations, as well as post accident mitigation actions. 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.

As described above, the proposed change provides only an alternative method of monitoring the position of a rod. No new accident initiators are introduced by the proposed alternative manner of performing rod position monitoring. The proposed change does not affect the reactor protection system or the reactor control system. Hence, no new failure modes are created that would cause a new or different kind of accident from any accident previously evaluated.

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 rod position indicators are required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. The proposed change does not alter the requirement to determine rod position but provides an alternative method for monitoring the position of the affected rod after the position of the rod is verified using the moveable incore detector system. As a result, the initial conditions of the accident analysis are preserved. The components affected by the alternate rod monitoring will not affect plant setpoints utilized for automatic mitigation of accident conditions or other equipment necessary for accident mitigation.

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

NRC Branch Chief: Michael L. Marshall, Jr.

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

Date of amendment request: July 12, 2006 (TS–06–03).

Description of amendment request: The proposed amendment would revise the limiting condition for operation for the Sequoyah Nuclear Plant, Units 1 and 2, Technical Specification (TS) Section 3.7.5, ''Ultimate Heat Sink.'' This revision would change the minimum ultimate heat sink (UHS) water elevation in TS 3.7.5.a from 670 feet to 674 feet. The essential raw cooling water (ERCW) temperature requirement in TS 3.7.5.b would be increased from 83 degrees Fahrenheit (°F) to 87 °F. The conditional requirements of TS 3.7.5.c would no longer be required and would be deleted by the proposed change. This change would also delete a footnote that established a temporary UHS temperature limit of 87 °F through September 30, 1995. These proposed changes are supported by a combination of design basis re-analysis, bounding analysis, and sensitivity analysis of the ERCW system, the UHS, and supported systems.

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 to increase the UHS maximum temperature and the minimum water level does not alter the function, design, or operating practices for plant systems or components. One exception is the elimination of non-safety-related station air compressor loads located in the turbine building. The UHS is utilized to remove heat loads from plant systems during normal and accident conditions. This function is not expected or postulated to result in the generation of any accident and continues to adequately satisfy the associated safety functions with the proposed changes. Therefore, the probability of an accident presently evaluated in the safety analyses will not be increased because the UHS function does not have the potential to be the source of an accident. The heat loads that the UHS is designed to accommodate have been evaluated for functionality with the higher temperature and elevation requirements. The result of these evaluations is that there is existing margins associated with the systems that utilize the UHS for normal and accident conditions. These margins are sufficient to accommodate the postulated normal and accident heat loads with the proposed changes to the UHS. Since the safety functions of the UHS are maintained, the systems that ensure acceptable offsite dose consequences will continue to operate as designed. 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 UHS function is not an initiator of any accident and only serves as a heat sink for normal and upset plant conditions. By allowing the proposed change in the UHS temperature and elevation requirements, only the parameters for UHS operation are changed while the safety functions of the UHS and systems that transfer the heat sink capability continue to be maintained. The UHS function provides accident mitigation capabilities and does not reflect the potential for accident generation. Therefore, the possibility for creating a new or different kind of accident is not created because the UHS is only utilized for heat removal functions that are not a potential source for accident generation. 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 has been evaluated for systems that are needed to support accident mitigation functions as well as normal operational evolutions. Operational margins were found to exist in the systems that utilize the UHS capabilities such that these proposed changes will not result in the loss of any safety function necessary for normal or accident conditions. The ERCW system has excess flow margins that will accommodate the increased flows necessary for the proposed temperature increase. While operating margins have been reduced by the proposed changes, safety margins have been maintained as assumed in the accident analyses for postulated events.

Additionally, the proposed changes do not require the modification of component setpoints utilized for automatic mitigation of accident conditions or other equipment necessary for accident mitigation. Therefore, a significant reduction in the margin to safety is not created by this proposed change. 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: L. Raghavan.

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

Date of amendment request: June 16, 2006 (WBN–TS–06–04).

Description of amendment request: The proposed amendment change would revise Technical Specification (TS) 5.7.2.11, ''Inservice Testing Program,'' to remove ''applicable supports'' from the Inservice Testing (IST) Program and revise the IST Program for pumps and valves to meet the requirements of the latest Edition and Addenda of the American Society of Mechanical Engineers (ASME) Code approved by the NRC for use on the date 12-months prior to the start of the 10-year IST Interval. For the Watts Bar Nuclear Plant (WBN), Unit 1, the second 10-year IST Interval will begin on December 27, 2006. The ASME Code that was approved in 10 CFR 50.55a(f)(4) for use on December 27, 2005, was ASME Operations and Maintenance (OM) Code, 2001 Edition, with Addenda through 2003. The proposed change provides consistency with the requirements in 10 CFR 50.55a(f)(4) by replacing the reference to ASME Boiler and Pressure Vessel Code, Section XI, with ASME OM Code. This proposed change is based on Technical Specification Task Force (TSTF) Traveler 479, Revision 0, ''Changes to Reflect Revision of 10 CFR 50.55a.'' TSTF 279–A, Revision 0, ''Remove 'applicable supports' from Inservice Testing Program,'' was approved by NRC and incorporated into Revision 2 of NUREG–1431, ''Standard Technical Specification Westinghouse Plants.'' In addition, the proposed amendment would add provisions to TS 5.7.2.11, Item b, to only apply Surveillance Requirement 3.0.2 to those IST frequencies of 2 years or less.

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 revises Technical Specification Section 5.7.2.11 for WBN Unit 1 to conform to the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves which are classified as ASME Code Class 1, 2, and 3.

ASME has in the last several years, transitioned the requirements for inservice testing of pumps and valves out of ASME Section XI and into a separate, stand alone code entitled the ''Code for Operation and Maintenance of Nuclear Power Plants,'' (ASME OM Code). The ASME OM Code has been endorsed by the NRC in 10 CFR 50.55a and is the Code that will be required for inservice testing of pumps and valves during the WBN Second Inservice Interval. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves. The proposed change also deletes the reference to supports from the Inservice Testing Program as supports are already inspected under the Inservice Inspection Program.

The proposed changes do not involve any hardware changes, nor do the changes affect the probability of any event initiators. There will be no change to normal plant operating parameters, accident mitigation capabilities, or accident analysis assumptions or inputs. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change revises the Technical Specifications to delete the reference to ''applicable supports'' from the Inservice Testing Program and to incorporate the latest Code requirements in 10 CFR 50.55a(f)(4) for Code Class 1, 2, and 3 pumps and valves for WBN's next ten year interval. The testing requirements are similar and reflect the same type testing. Valves are still stroke timed; remote position indicators are still verified to be accurate; seat leakage measurements of critical valves are still performed; relief valves still have their setpoints and seat leakages verified; pumps are still tested for hydraulic performance and mechanical condition; check valves are verified to open and close properly; and supports are still inspected under the appropriate inspection program.

The proposed changes do not involve a modification to the physical configuration of the plant or change methods governing normal plant operation. No test methods are added or deleted. 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 revises the TS for consistency with the Standard Technical Specification and with the requirements in 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves which are classified as ASME Code Class 1, 2, and 3. This change incorporates revisions to the ASME Code that result in a net improvement in the measures of testing. Incorporation of the ASME OM Code does not alter the limiting values and acceptance criteria used to judge the continued acceptability of components tested by the Inservice Testing Program. Deletion of the reference to supports in the Inservice Testing Program does not alter the support inspection program as the program is currently under the Inservice Inspection Program. Since these limits are not altered, the margin of safety is not altered. 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: L. Raghavan.

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

Date of amendment request: May 30, 2006.

Description of amendment request: The amendment would revise Surveillance Requirements (SRs) 3.5.2.8 and 3.6.7.1 in the Technical Specifications (TSs), and delete the footnote to the frequency for SR 3.5.2.5. SR 3.5.2.8 would be revised by replacing the phrase ''trash racks and screens'' with the word ''strainers.'' This reflects (1) the replacement of the existing containment recirculation sump suction inlet trash racks and screens with strainers with significantly greater effective surface area, and (2) the resulting relocation of the recirculation fluid pH control system in Refueling Outage 15 schedule for the spring of 2007. The footnote to SR 3.5.2.5 would be deleted because it is no longer applicable to the TSs.

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

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

Response: No.

None of the changes impact the initiation or probability of occurrence of any accident [previously evaluated].

The consequences of accidents evaluated in the FSAR [Final Safety Analysis Report for the Callaway Plant] that could be affected by this proposed change are those involving the pressurization of the containment and associated flooding of the containment and recirculation of this fluid within the Emergency Core Cooling System (ECCS) or the Containment Spray System (CSS) (e.g., LOCAs [Loss-of-Coolant Accidents]). [The containment sump trash racks and screens, and the sump strainers that are replacing the trash racks and screens are not initiators of accidents.]

Although the configurations of the existing sump screen and the replacement strainer assemblies are different, they serve the same fundamental purpose of passively removing debris from the suction of the supported system pumps. Removal of trash racks does not impact the adequacy of the pump NPSH [net positive suction head] assumed in the safety analyses. Likewise the change does not reduce the reliability of any supported systems or introduce any new system interactions. The greatly increased surface area of the new strainer is designed to reduce head loss [at the containment sump] and reduce the approach velocity at the strainer face significantly, decreasing the risk of impact from large debris entrained in the sump flow stream.

The recirculation fluid pH control system storage baskets serve a passive function to provide a buffering agent to neutralize the sump solution. The redesign and relocation of the storage baskets are considered a like kind replacement. The baskets will be located within the flood plain and will continue to ensure that the buffering agent is dissolved in the sump fluid to ensure an equilibrium pH $\geq$ 7.1. Failure of a basket would not initiate an accident. The ECCS and CSS will continue to function in a manner consistent with the plant design basis.

As such, the proposed change to the Technical Specifications Surveillance Requirements does not involve a significant increase in the probability or consequences of an accident previously evaluated. The installed quantity of trisodium phosphate Crystalline will provide a minimum equilibrium sump pH of 7.1 following dissolution and mixing. [Deleting the footnote to SR 3.5.2.5 is an administrative change to remove a one-time required verification that has already been performed and is no longer a requirement in the current TSs.] Therefore, there is not a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The containment recirculation sump strainers and recirculation fluid pH control system are passive systems used for accident mitigation. As such, they cannot be accident initiators. Therefore, there is no possibility that this change could create any accident of any kind. [The containment recirculation sump suction inlet trash racks and screens are being replaced with a complex strainer design with significantly larger effective surface area to reduce head loss and reduce the approach velocity at the strainer face significantly, decreasing the risk of impact from large debris entrained in the sump flow stream. This will result in the recirculation fluid pH control system being relocated.]

No new accident scenarios, transient precursors, or limiting single failures are introduced as a result of these changes. There will be no adverse effect[s] or challenges imposed on any safety-related system as a result of these changes. The quantity of trisodium phosphate crystalline will provide a minimum equilibrium sump pH of ≥ 7.1 following dissolution and mixing. Therefore, the possibility of a new or different type of accident is not created.

There are no changes which would cause the malfunction of safety-related equipment, assumed to be operable in the accident analyses, as a result of the proposed Technical Specification changes. No new equipment performance burdens are imposed. The possibility of a malfunction of safety-related equipment with a different result is not created. [Deleting the footnote to SR 3.5.2.5 is an administrative change to remove a one-time required verification that has already been performed and is no longer a requirement in the current TSs.] Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do[es] the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed changes do not adversely affect any plant safety limits, setpoints, or design parameters. The changes also do not adversely affect the fuel, fuel cladding, Reactor Coolant System (RCS), or containment integrity. [The radiological dose consequence acceptance criteria in the Standard Review Plan for accidents will continue to be met. Deleting the footnote to SR 3.5.2.5 is an administrative change to remove a one-time required verification that has already been performed and is no longer a requirement in the current TSs.] Therefore, the proposed TS 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: John O'Neill, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, NW., Washington, DC 20037.

NRC Branch Chief: David Terao.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: May 30, 2006, as supplemented by letter dated June 30, 2006.

Description of amendment request: The proposed amendments would relocate the American Society for Testing and Materials (ASTM) standard being used to test the total particulate concentration of the stored fuel oil to the TS Bases. This proposed change is described in TS Task force (TSTF) Standard TS Change Traveler TSTF–374–A, Rev. 0, ''Revision to TS 5.5.13 and Associated TS Bases for Diesel Fuel Oil.'' In addition, the licensee has proposed to use a ''water and sediment test'' instead of the ''clear and bright'' test provided in TSTF–374.

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

The proposed change relocates the specific ASTM reference from the Administrative Controls Section of Technical Specifications (TS) to a licensee-controlled document. Relocating the specific ASTM Standard reference from the TS to a licensee-controlled document will not affect nor degrade the ability of the EDGs [emergency diesel generators] to perform their specified safety function. Fuel oil quality will continue to meet the current ASTM requirements for particulate concentration.

The proposed change is administrative in nature and does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed change does not alter or prevent the ability of structures, systems or components from performing their intended function to mitigate the consequences on an initiating event with the assumed acceptance limits. The proposed change does 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 change does not increase the types and amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposure.

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

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

The proposed change relocates the specific ASTM reference from the Administrative Controls Section of Technical Specifications to a licensee-controlled document.

The change does not involve a physical alteration of the plant or a change in the methods governing normal plant conditions. In addition, the change does not impose any new or different requirements or eliminate any existing requirements. The change does not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change relocates the specific ASTM reference from the Administrative Controls Section of TS to a licensee-controlled document. The detail associated with the specific ASTM standard reference is not required to be in the TS to provide adequate protection of the public health and safety, since the TS still retain the requirement for compliance with the applicable ASTM standard.

The level of safety of facility operation is unaffected by the proposed change since there is no change in the intent of the TS requirements of assuring fuel oil is of the appropriate quality for EDG use. The proposed change provides the flexibility needed to maintain state-of-the-art technology in fuel oil sampling and analysis methodology.

The proposed change does not reduce a margin of safety since it has no impact on any transient or safety.

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Counsel, Dominion Resources Services, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385.

NRC Branch Chief: Evangelos C. Marinos.

Virginia Electric and Power Company, Docket Nos. 50–280 and 50–281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia

Date of amendment request: May 26, 2006.

Description of amendment request: Item 1: The proposed amendments would revise the Technical Specification (TS) requirements related to Reactor Coolant System (RCS) leakage definitions and requirements and steam generator tube integrity. The licensee requested this change to implement TS Task Force (TSTF) Standard TS Change Traveler, TSTF–449, ''Steam Generator

Tube Integrity,'' (TSTF–449, Rev. 4). Item 2: In addition, in its submittal dated May 26, 2006, the licensee proposed minor deviations from the TS changes described in TSTF–449, Rev. 4, to provide consistency with Surry's custom TSs.

Basis for proposed no significant hazards consideration determination: Item 1: As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration 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 requires a SG Program that includes performance criteria that will provide reasonable assurance that the SG tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational leakage.

A SG tube rupture (TR) event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. In the analysis of a SGTR event, a bounding primary to secondary leakage rate equal to the operational leakage rate limits in the licensing basis plus the leakage rate associated with a double-ended rupture of a single tube is assumed.

For other design basis accidents such as main steam line break (MSLB), rod ejection, and reactor coolant pump locked rotor the tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). These analyses typically assume that primary to secondary leakage for all SGs is 1 gallon per minute or increases to 1 gallon per minute as a result of accident induced stresses. The accident induced leakage criterion introduced by the proposed changes accounts for tubes that may leak during design basis accidents. The accident induced leakage criterion limits this leakage to no more than the value assumed in the accident analysis.

The SG performance criteria proposed change to the TS identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the SG Program required by the proposed change to the TS. The program, defined by NEI 97–06, Steam Generator Program Guidelines, includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring. The proposed changes do not, therefore, significantly increase the probability of an accident previously evaluated.

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT 1–131 in the primary coolant and the primary to secondary leakage rates resulting from an accident. Therefore, limits are included in the plant TS for operational leakage and for DOSE EQUIVALENT 1–131 in primary coolant to ensure the plant is operated within its analyzed condition. The typical analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gallon per minute with no more than 500 gallons per day in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT 1–131 are at the TS values before the accident.

The proposed change does not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed approach updates the current TSs and enhances the requirements for SG inspections. The proposed change does not adversely impact any other previously evaluated design basis accident and is an improvement over the current TSs.

Therefore, the proposed change does not affect the consequences of a SGTR accident and the probability of such an accident is reduced. In addition, the proposed changes do not affect the consequences of an MSLB, rod ejection, or a reactor coolant pump locked rotor event, or other previously evaluated accident.

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

The proposed performance based requirements are an improvement over the requirements imposed by the current [TS]. Implementation of the proposed SG Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The result of the implementation of the SG Program will be an enhancement of SG tube performance. Primary to secondary leakage that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. In addition, the proposed change does not impact any other plant system or component. The change enhances SG inspection requirements.

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

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

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of an SG is maintained by ensuring the integrity of its tubes.

[SG] tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the SG Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the SG Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TSs.

For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by the proposed change to the TS.

The NRC staff has reviewed the licensee's incorporation of the above analysis by reference 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.

Item 2: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below.

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

The proposed changes involve adding a new definition for RCS [reactor coolant system] leakage and rewording certain [TSs] for consistency with NUREG–1431, Revision 3. These changes do not involve any physical plant modifications or changes in plant operation; consequently, no technical changes are being made to the existing TS. As such, these changes are administrative in nature and do not affect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes involve adding a new definition for RCS leakage and rewording certain [TSs] for consistency with NUREG–1431, Revision 3. These administrative changes do not involve physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes involve adding a new definition for RCS leakage and rewording certain [TS] for consistency with NUREG–1431, Revision 3. The changes are administrative in nature and will not involve any technical changes. The changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. Also, since these changes are administrative in nature, no question of safety is involved. 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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Counsel, Dominion Resources Services, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385.

NRC Branch Chief: Evangelos C. Marinos.

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.*

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

Date of application for amendments: May 26, 2005, as supplemented by letters dated May 23 and June 20, 2006.

Brief description of amendments: The amendments revised Technical Specification (TS) 1.1, ''Definitions,'' TS 3.4.14, ''RCS [reactor coolant system] Operational Leakage,'' TS 5.5.9, ''Steam Generator (SG) Program,'' and TS 5.6.8, ''Steam Generator Tube Inspection Report,'' and added a new specification, TS 3.4.18, ''Steam Generator (SG) Tube Integrity.'' The changes are consistent with TS Task Force (TSTF) Change TSTF–449, Revision 4, ''Steam Generator Tube Integrity.''

Date of issuance: July 27, 2006.

Effective date: As of the date of issuance to be implemented within 150 days from the date of issuance.

Amendment Nos.: Unit 1–161, Unit 2–161, Unit 3–161.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Operating Licenses and the Technical Specifications for all three units.

*Date of initial notice in **Federal Register:*** July 5, 2005 (70 FR 38714). The May 23 and June 20, 2006, supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 27, 2006.

No significant hazards consideration comments received: No.

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2), Darlington County, South Carolina

Date of application for amendment: January 21, 2005, as supplemented by letters dated May 26, 2005, September 19, 2005, and March 31, 2006.

Brief description of amendment: The amendment approves the implementation of the alternative source term methodology for a loss-of-coolant accident at HBRSEP2.

Date of issuance: July 11, 2006.

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

Amendment No. 207.

Renewed Facility Operating License No. DPR–23. Amendment does not revise the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 24, 2005 (70 FR 29786). The supplemental letters dated May 26, 2005, September 19, 2005, and March 31, 2006, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 11, 2006.

No significant hazards consideration comments received: No.

Dominion Energy Kewaunee, Inc. Docket No. 50–305, Kewaunee Power Station, Kewaunee County, Wisconsin

Date of application for amendment: January 12, 2006, as supplemented by letter dated June 2, 2006.

Brief description of amendment: The amendment revises the existing steam generator (SG) tube surveillance program to be consistent with TS Task Force (TSTF) Change TSTF–449, Revision 4, ''Steam Generator Tube Integrity,'' and the model safety evaluation prepared by the Nuclear Regulatory Commission (NRC) and published in the **Federal Register** on March 2, 2005 (70 FR 10298) under the consolidated line item improvement process (CLIIP).

Date of issuance: July 18, 2006.

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

Amendment No.: 188.

Facility Operating License No. DPR–43: Amendment revised the Facility Operating License and Technical Specifications.

*Date of initial notice in **Federal Register:*** February 14, 2006 (71 FR 7806). The supplement letter contained clarifying information and did not change the initial no significant hazards consideration determination, and did not expand the scope of the orginal **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 18, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 27, 2004.

Brief description of amendments: The amendments revised the facility operating licenses by removal of license condition 2.F, ''Reporting Requirements'', with regard to maximum power level, Updated Final Safety Analysis Report, antitrust conditions, fire protection, and additional conditions.

Date of issuance: July 31, 2006.

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

Amendment Nos.: 230, 226.

Renewed Facility Operating License Nos. NPF–35 and NPF–52: Amendments revised the licenses.

*Date of initial notice in **Federal Register:*** July 5, 2005 (70 FR 38717).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 31, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 17, 2006.

Brief description of amendment: The amendment allows a delay time for entering a supported system Technical Specification (TS) when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed consistent with the program in place for complying with the requirements of 10 CFR 50.65(a)(4). Limiting Condition for Operation (LCO) 3.0.8 is added to the TS to provide this allowance and define the requirements and limitations for its use.

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–372, Revision 4. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on November 24, 2004 (69 FR 68412), on possible amendments concerning TSTF–372, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 4, 2005 (70 FR 23252). The licensee affirmed the applicability of the following NSHC determination in its application dated April 17, 2006.

Date of issuance: July 11, 2006.

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

Amendment No.: 198.

Facility Operating License No. NPF–21: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 9, 2006 (71 FR 26998).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 11, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 26, 2005, as supplemented by letter dated April 11, 2006.

Brief description of amendment: The amendment revises the analysis method used for the large-break loss-of-coolant accident.

Date of issuance: July 24, 2006.

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

Amendment No.: 248.

Facility Operating License No. DPR–26: The amendment revised the Technical Specifications and License.

*Date of initial notice in **Federal Register:*** November 8, 2005 (70 FR 67747). The April 11, 2006, supplement provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 24, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 26, 2006, as supplemented by letter dated April 12, 2006.

Brief description of amendment: The amendment approves the implementation of the Boiling Water Reactor Vessel and Internals Project reactor pressure vessel integrated surveillance program as the basis for demonstrating the compliance of JAFNPP with the requirements of Appendix H to Title 10 of the Code of Federal Regulations part 50.

Date of issuance: July 26, 2006.

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

Amendment No.: 285.

Facility Operating License No. DPR–59: The amendment revised the Updated Final Safety Analysis Report and the License.

*Date of initial notice in **Federal Register:*** March 14, 2006 (71 FR 13174). The April 12, 2006, supplement provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 26, 2006.

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 application for amendment: September 19, 2005.

Brief description of amendment: The amendment modified ANO–2 Surveillance Requirement TS 3.1.1.4, ''Moderator Temperature Coefficient,'' and allowed the use of WCAP–16011–P–A, ''Startup Test Activity Reduction Program.''

Date of issuance: August 2, 2006.

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

Amendment No.: 265.

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

*Date of initial notice in **Federal Register:*** December 6, 2005 (70 FR 72671).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 2, 2006.

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 application for amendment: September 19, 2005, as supplemented by letters dated May 11 and June 19, 2006.

Brief description of amendment: The amendment revised the existing steam generator tube surveillance program to be consistent with the U.S. Nuclear Regulatory Commission's approved Technical Specification Task Force Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity,'' Revision 4. TSTF–449 is part of the consolidated line item improvement process.

Date of issuance: August 2, 2006.

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

Amendment No.: 266.

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

Date of initial notice in ***Federal Register:*** January 3, 2006 (71 FR 147). The supplements dated May 11 and June 19, 2006, 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 amendment is contained in a Safety Evaluation dated August 2, 2006.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: January 25, 2006, as supplemented by letter dated May 17, 2006.

Brief description of amendments: The amendment revised the Quad Cities licensing basis, as described in the Updated Final Safety Analysis Report, to allow the use of automatic load tap changers to operate in automatic mode on the reserve auxiliary transformers to compensate for potential offsite power voltage fluctuations, in order to ensure that acceptable voltage is maintained for safety-related equipment.

Date of issuance: July 24, 2006.

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

Amendment Nos.: 232 and 228.

Renewed Facility Operating License Nos. DPR–29 and DPR–30: The amendments revised the License.

Date of initial notice in ***Federal Register:*** May 23, 2006 (71 FR 29678). The May 17, 2006, supplement contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 24, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 23, 2005, as supplemented on April 6, 2006.

Brief description of amendments: The amendments extended the licensed lives of the Diablo Canyon Power Plant, Unit Nos. 1 and 2 reactors by the amount of time the licensee had expended to perform low-power testing of the reactors prior to initial startup.

Date of issuance: July 17, 2006.

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

Amendment Nos.: Unit 1–188; Unit 2–190.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Facility Operating Licenses.

Date of initial notice in ***Federal Register:*** October 11, 2005 (70 FR 59087). The April 6, 2006, supplemental letter provided additional information that clarified the application, and did not expand the scope of the application as originally noticed.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 17, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 4, 2005, as supplemented by letters dated February 9, July 18, and August 1, 2006.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.7.1.3, ''Ultimate Heat Sink,'' to permit continued plant operation if the temperature of the ultimate heat sink (UHS) exceeds 89 °F, provided the UHS temperature averaged over the previous 24-hour period is verified at least once per hour to be less than or equal to 89 °F, and the UHS temperature does not exceed a maximum value of 91.4 °F.

Date of issuance: August 1, 2006.

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

Amendment No.: 168.

Facility Operating License No. NPF–57: This amendment revised the TSs.

Date of initial notice in ***Federal Register:*** August 30, 2005 (70 FR 51382).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 1, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 7, 2005, as supplemented on May 5, 2006.

Brief description of amendment: The amendment revises Technical Specification 3.9.3, ''Containment Penetrations,'' to allow an emergency egress door, access door, or roll up door, as associated with the equipment hatch penetration, to be open, but capable of being closed, during core alterations or movement of irradiated fuel within containment.

Date of issuance: July 26, 2006.

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

Amendment No.: 98.

Renewed Facility Operating License No. DPR–18: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 3, 2006 (71 FR 154). The May 5, 2006, letter 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 amendment is contained in a Safety Evaluation dated July 26, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 18, 2005.

Brief description of amendment: The amendment revises the frequency in Technical Specification Surveillance Requirement 3.6.6.15, which verifies that each containment spray nozzle is unobstructed. The frequency is changed from ''10 years'' to ''following maintenance which could result in nozzle blockage.''

Date of issuance: July 31, 2006.

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

Amendment No.: 99.

Renewed Facility Operating License No. DPR–18: Amendment revised the Technical Specifications and the License.

*Date of initial notice in **Federal Register:*** January 3, 2006 (71 FR 154).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 31, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 6, 2004 (TS 428) as supplemented by letter dated June 16, 2005.

Brief description of amendment: The amendment revised the reactor vessel Pressure-Temperature curves depicted in the Technical Specification (TS) Figure 3.4.9–1 and adds a new TS Figure 3.4.9–2.

Date of issuance: July 26, 2006.

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

Amendment No.: 256.

Facility Operating License No. DPR–33: Amendment revised the TS.

*Date of initial notice in **Federal Register:*** January 18, 2005 (70 FR 2899). The supplement dated June 16, 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 amendment is contained in a Safety Evaluation dated July 26, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 15, 2005 (TS–05–09), as supplemented by letter dated June 7, 2006.

Brief description of amendment: The amendment revises the Watts Bar Nuclear Plant (WBN) Technical Specification Surveillance Requirements to increase the minimum required average ice basket weight, thus, increasing the corresponding total weight of the stored ice in the WBN ice condenser. The changes to the ice basket and total ice weights are due to the additional energy associated with the Replacement Steam Generators.

Date of issuance: July 25, 2006.

Effective date: As of the date of issuance and shall be implemented prior to Mode 4 at startup to begin Cycle 8 fuel cycle.

Amendment No. 62.

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

*Date of initial notice in **Federal Register:*** February 14, 2006 (71 FR 7814). The supplemental letter provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 25, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 28, 2006.

Brief description of amendment: The amendment revised Technical Specification 5.0, ''Administrative Controls,'' by changing a position title and department name.

Date of issuance: July 11, 2006.

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

Amendment No.: 173.

Facility Operating License No. NPF–30: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 9, 2006 (71 FR 27005).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 11, 2006.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units 1 and 2, Louisa County, Virginia

Date of application for amendment: July 5, 2005, as supplemented by letters dated March 30, April 13, and May 11, 2006.

Brief description of amendment: The amendments revised the Technical Specifications (TSs) to add a reference in TS 5.65.b, ''Core Operating Limits Report (COLR),'' to permit the use of an alternate methodology to perform a thermal-hydraulic analysis to predict the critical heat flux and departure from nucleate boiling ratio for the AREVA Advanced Mark-BW fuel in the North Anna 1 and 2 cores.

Date of issuance: July 21, 2006.

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

Amendment Nos.: 247, 227.

Renewed Facility Operating License Nos. NPF–4 and NPF–7: Amendments changed the Licenses and the TSs.

*Date of initial notice in **Federal Register:*** August 16, 2005 (70 FR 48208). The supplements dated March 30, April 13, and May 11, 2006, contained clarifying information only and did not change the initial no significant hazards consideration determination or expand the scope of the initial application.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 21, 2006.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 8th day of August, 2006.

For the Nuclear Regulatory Commission.

Catherine Haney,

Director, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation.

[FR Doc. 06–6921 Filed 8–14–06; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–54296; File No. SR–ISE–2006–30]

Self-Regulatory Organizations; International Securities Exchange, Inc.; Order Approving a Proposed Rule Change, and Amendment No. 1 Thereto, Increasing the Linkage Inbound Principal Order Fee

August 9, 2006.

On June 5, 2006, the International Securities Exchange, Inc. (''ISE'' or ''Exchange'') filed with the Securities and Exchange Commission (''Commission''), pursuant to Section 19(b)(1) of the Securities Exchange Act of 1934 (''Act'') [1] and Rule 19b–4 thereunder,[2] a proposed rule change to amend its Schedule of Fees in the manner described below. On June 29,

[1] 15 U.S.C. 78s(b)(1).

[2] 17 CFR 240.19b–4.