For The Nuclear Regulatory Commission
Joseph M. Sebrosky,
Senior Project Manager, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards.
[FR Doc. E5–2586 Filed 5–23–05; 8:45 am]
BILLING CODE 7590–01–P

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 April 29, 2005 through May 12, 2005. The last biweekly notice was published on May 10, 2005 (70 FR 24645).

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.*

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

Date of amendment request: January 21, 2005.

Description of amendment request: The proposed amendment would implement the Alternative Source Term (AST) for the analysis of the radiological consequences of a design-basis Loss-of-Coolant Accident (LOCA). There are no changes proposed to the Operating License or Technical Specifications.

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

Revision of the LOCA analysis to the Alternative Source Term methodology does not affect the design or operation of HBRSEP [H. B. Robinson Steam Electric Plant], Unit No. 2. Rather, once the occurrence of an accident has been postulated, the new source term is an input to evaluate the consequences of the postulated accident. The implementation of the Alternative Source Term has been evaluated in revisions to the LOCA dose analysis at HBRSEP, Unit No. 2. Based on the results of this analysis, it has been demonstrated that the dose consequences are within the regulatory guidance provided by the NRC. This guidance is presented in 10 CFR 50.67 and Regulatory Guide 1.183.

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

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

The proposed change does not affect plant structures, systems, or components. The proposed change is to an evaluation methodology and does not initiate design basis accidents.

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

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

The proposed change is associated with the implementation of a new licensing basis for HBRSEP, Unit No. 2. The new licensing basis implements an Alternative Source Term in accordance with 10 CFR 50.67 and the associated Regulatory Guide 1.183. The results of the revised limiting design basis analysis are subject to revised acceptance criteria. The analysis has been performed using conservative methodologies in accordance with regulatory guidance or other methodologies approved by the NRC in prior plant-specific license amendments. The dose consequences are within the acceptance criteria found in the regulatory guidance associated with Alternative Source Terms.

The proposed change continues to ensure that doses at the exclusion area and low population zone boundaries, as well as the control room, are within the corresponding regulatory limits. Specifically, the margin of safety for the radiological consequences of these accidents is considered to be that provided by meeting the applicable regulatory limits.

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

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

Attorney for licensee: David T. Conley, Associate General Counsel II—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Michael L. Marshall, Jr.

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

Date of amendment request: February 14, 2005.

Description of amendment request: The proposed amendment would revise the surveillance requirements (SRs) for the station batteries as specified in Technical Specification (TS) SR 3.8.4.5, the battery service test, and TS SR 3.8.4.6, the battery performance test.

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

1. Do the Proposed Changes Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated?

No. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed surveillance changes will continue to ensure that the DC system is tested in a manner that will verify operability. Performance of the required system surveillances, in conjunction with the applicable operational and design requirements for the DC system, provide assurance that the system will be capable of performing the required design functions for accident mitigation and also that the system will perform in accordance with the functional requirements for the system as described in the Updated Final Safety Analysis Report for HBRSEP [H. B. Robinson Steam Electric Plant], Unit No. 2. This ensures that the rate of occurrence and consequences of analyzed accidents will not change. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the Proposed Changes Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated?

No. The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated. The proposed surveillance requirement changes will continue to ensure that the DC system is tested in a manner that will verify operability. No physical changes to the HBRSEP, Unit No. 2, systems, structures, or components are being implemented. There are no new or different accident initiators or sequences being created by the proposed Technical Specifications changes. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the Proposed Changes Involve a Significant Reduction in the Margin of Safety?

No. The proposed changes do not involve a significant reduction in the margin of safety. The proposed DC system surveillance requirement changes provide appropriate and applicable surveillances for the DC system. The proposed changes to surveillance requirements for the DC system will continue to ensure system operability. Therefore, these changes do not affect any margin of safety for HBRSEP, Unit No. 2.

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 T. Conley, Associate General Counsel II—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Michael L. Marshall, Jr.

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

Date of amendment request: March 3, 2005.

Description of amendment request: The proposed amendment would revise the requirements of Technical Specification (TS) 5.6.5, ''Core Operating Limits Report (COLR).'' Specifically, the proposed change would add topical report EMF–2103(P)(A), ''Realistic Large Break LOCA [loss-of-coolant accident] Methodology for Pressurized Water Reactors,'' to the list of documents specified in TS 5.6.5. TS 5.6.5 lists the approved methodologies that can be used to determine the core operating limits.

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 methodology will be reviewed and approved by the NRC prior to its use for HBRSEP [H. B. Robinson Steam Electric Plant], Unit No. 2. Analyzed events are assumed to be initiated by the failure of plant structures, systems, or components. The determination of core operating limits in accordance with this new methodology will meet the limitations specified in the NRC safety evaluation of the new methodology. The topical report associated with the new methodology demonstrates that the integrity of the fuel will be maintained and that design requirements will continue to be met. The proposed change does not involve physical changes to any plant structure, system, or component. Therefore, the probability of occurrence for a previously analyzed accident is not significantly increased.

The consequences of a previously analyzed accident are dependent on the initial conditions assumed for the analysis, the behavior of the fuel during the analyzed accident, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. The proposed methodology continues to meet applicable design and safety analyses acceptance criteria. The proposed change does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. As a result, no analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident. The proposed change does not affect setpoints that initiate protective or mitigative actions. The proposed change ensures that plant structures, systems, or components are maintained consistent with the safety analysis and licensing bases. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve any physical alteration of plant systems, structures, or components, other than allowing for fuel design in accordance with NRC approved methodologies. The proposed methodology continues to meet applicable criteria for Large Break Loss of Coolant Accident (LBLOCA) analysis. No new or different equipment is being installed. No installed equipment is being operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints that initiate protective or mitigative actions. As a result, no new failure modes are being introduced. There are no changes in the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety is established through the design of the plant structures, systems, and components, through the parameters within which the plant is operated, through the establishment of the setpoints for the actuation of equipment relied upon to respond to an event, and through margins contained within the safety analyses. The proposed change in the methodology used for LBLOCA analyses does not impact the condition or performance of structures, systems, setpoints, and components relied upon for accident mitigation. The proposed change does not significantly impact any safety analysis assumptions or results. Therefore, the proposed change does not result in a significant reduction in the margin of safety.

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

Attorney for licensee: David T. Conley, Associate General Counsel II—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Michael L. Marshall, Jr.

Dominion Nuclear Connecticut, Inc., Docket Nos. 50–245, 50–336, and 50–423, Millstone Power Station, Unit Nos. 1, 2, and 3, New London County, Connecticut

Date of amendment request: December 21, 2004.

Description of amendment request: The requested change will delete Technical Specification (TS) requirements for annual Occupational Radiation Exposure Reports (all units), annual report regarding challenges to pressurizer relief and safety valves (Units 2 and 3), and Monthly Operating Reports (Units 2 and 3).

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing license amendment applications in the **Federal Register** on June 23, 2004 (69 FR 35067). The licensee affirmed the applicability of the model NSHC determination in its application dated December 21, 2004.

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

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

Response: No.

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

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

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

Response: No.

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

Based upon the reasoning presented above, the requested change does not involve significance hazards consideration.

Attorney for licensee: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

NRC Section Chief: Darrell J. Roberts.

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

Date of amendment request: March 9, 2005.

Description of amendment request: Current Technical Specifications (TSs) require that all operations involving a reduction in reactor coolant boron concentration or that involve positive reactivity changes be suspended under certain conditions. The requested changes modify the TSs to incorporate wording related to the reactor coolant system (RCS), electrical power systems, and refueling operations to provide operational flexibility during mode changes or addition of coolant during shutdown operations. Additionally, changes are to be made to the TS bases, as appropriate.

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

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

Response: No.

The proposed change does not in any way alter the SDM [shutdown margin] or refueling boron concentration. It limits introduction of coolant into the RCS of reactivity more positive than that necessary to meet the required SDM or refueling boron concentration. This proposed change does not affect the input or assumptions for any accidents previously evaluated nor does it affect initiation of an accident. Based on this discussion, the proposed amendment does not increase the probability or consequence of an accident previously evaluated.

Criterion 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 allows introduction of coolant into the RCS with different temperature or lower boron concentration, however, the required boron concentration or SDM is maintained. The proposed amendment does not introduce failure modes, accident initiators, or malfunctions that would cause a new or different kind of accident. No plant modifications are associated with the change. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change provides the flexibility necessary for continued safe reactor operations while limiting any potential for excess positive reactivity additions. [The] SDM and required boron concentration are not affected. Therefore, based on the above, 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: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Waterford, CT 06141–5127.

NRC Section Chief: Darrell J. Roberts.

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

Date of amendment request: December 23, 2004.

Description of amendment request: The requested amendment would relocate certain Technical Specifications regarding refueling operations to the Technical Requirements Manual (TRM).

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

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

Response: No.

The communications equipment, refueling machine, and spent fuel pool crane are not designed to perform accident mitigation functions. The proposed change to relocate selected refueling specifications does not modify any plant equipment and does not impact any failure modes that could lead to an accident. Relocating the specifications to the TRM where changes would be controlled under the 10 CFR 50.59 process does not change the ability of the communications or refueling equipment to function as expected. Additionally, these specifications have no affect on the consequence of any analyzed accident since the equipment is not related to accident mitigation. Based on this discussion, the proposed amendment does not increase the probability or consequences of an accident previously evaluated.

Criterion 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[s] do[es] not modify any plant equipment and there is no impact on the capability of the existing equipment to perform their intended functions to move fuel safely or conduct refueling operations while in contact with the control room. No system setpoints are being modified and no changes are being made to the method in which refueling operations are conducted. No changes to the heavy loads program are being proposed by this change. No new failure modes are introduced by the proposed changes. The proposed amendment does not introduce accident initiators or malfunctions that would cause a new or different kind of accident. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The relocation of Technical Specification 3/4.9.5, ''Refueling Operations, Communications,'' to the TRM does not imply any reduction in its importance in [e]nsuring communication between the control room and the refueling station. The proposed change will not alter the requirement on communication between the control room and the refueling station, it will not alter any of the assumptions used in the fuel handling accident analysis, nor will it cause any safety system parameters to exceed their acceptance limit. The relocation of Technical Specification 3/4.9.6, ''Refueling Machine'' to the TRM does not alter the requirement for the lifting device on the refueling machine to have adequate capacity or for the interlocks to be demonstrated operable prior to fuel movement. The assumptions used in the accident analysis are not impacted by this change and no impact to any safety system parameters will result. The relocation of Technical Specification 3/4.9.7, ''Crane Travel—Spent Fuel Storage Areas,'' to the TRM will not alter the requirement that the crane interlocks and/or physical stops are operable, nor will it alter any of the assumptions used in the fuel handling accident analysis. Heavy load lifts are administratively controlled by a safe load path and crane interlocks. The proposed change[s] do[es] not modify any heavy load path criteria. Administrative changes associated with the proposed revision such as relocation of associated Technical Specification Bases to the TRM will not have an impact on any established safety margins.

The proposed change[s] do[es] not affect any of the assumptions used in the accident analysis, nor do they affect any operability requirements for equipment important to plant safety. Therefore, the proposed change[s] will not result in a significant reduction in the margin of safety as defined in the Bases for Technical Specifications covered in this License Amendment Request.

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: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Waterford, CT 06141–5127.

NRC Section Chief: Darrell J. Roberts.

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

Date of amendment request: November 25, 2002, as supplemented by letters dated November 13, and December 16, 2003, September 22, 2004, and April 6, 2005.

Description of amendment request: The amendments would revise the Technical Specifications (TS) for the Ventilation Filter Testing Program (VFTP), Annulus Ventilation System (AVS), Auxiliary Building Filtered Ventilation Exhaust System (ABFVES), Fuel Handling Ventilation Exhaust System (FHVES), and Control Room Area Ventilation System (CRAVS), and containment penetrations.

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:

First Standard

Does operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated? No.

This license amendment request proposes amendments to the system TS and/or Bases and/or VFTP TS requirements for the AVS, ABFVES, FHVES, and CRAVS. It also proposes amendments to the TS and Bases for Containment Penetrations. The AVS is in standby during normal plant operations and operates only following a Safety Injection signal or during a test. It is not an accident initiator. The ABFVES is in operation during normal plant operations. However, the ABFVES is not used in direct support of any phase of power generation or conversion or transmission, shutdown cooling, fuel handling operations, or processing of radioactive fluids. Therefore, it is not an accident initiator. The FHVES is utilized to support fuel handling operations when moving recently irradiated fuel. It is not an accident initiator. The CRAVS operates during normal plant operations. However, it is not an accident initiator (the CRAVS being defined so as to exclude equipment that maintains an appropriately low temperature in the control room). The status of containment penetrations is required to be controlled so as to minimize the consequences of a fuel handling accident or a weir gate drop accident. The containment penetrations by themselves are not accident initiators. No accident initiators are associated with the changes proposed in this license amendment request. For these reasons, operation of the facility in accordance with this proposed amendment does not involve a significant increase in the probability of any accident previously evaluated.

In support of the proposed amendment, an analysis has been performed to determine the radiological consequences of the design basis [Loss of Coolant Accident] LOCA at Catawba Nuclear Station. The analysis made use of the Alternative Source Term (AST) methodology and in general conformed to the regulatory positions of Regulatory Guide 1.183 and the draft regulatory positions of DG–1111. Total Effective Dose Equivalent (TEDE) radiation doses at the Exclusion Area Boundary (EAB), boundary of the Low Population Zone (LPZ), and to the control room operators were calculated and found to be acceptable. TEDEs were calculated for a design basis LOCA postulated for a Catawba nuclear unit operating with all low enriched uranium (LEU) fuel and with 4 mixed oxide (MOX) lead fuel assemblies (LFAs). It was found that insertion of 4 MOX LFAs did not produce a significant increase in the TEDEs for a design basis LOCA.

* * * * *

The new value for the control room TEDE radiation dose is higher than the TEDE radiation dose equivalent to the radiation doses currently reported in the UFSAR. However, the limiting control room TEDE radiation dose reported in this submittal is lower than the acceptance criterion * * * The new LPZ TEDE radiation dose is higher than the equivalent TEDE radiation dose currently represented. On the other hand, the margin to the acceptance criterion is [large] * * *. The TEDE radiation doses newly computed at the EAB for the design basis LOCA are lower than the corresponding equivalent EAB TEDE radiation dose currently represented in the UFSAR. The margin in the EAB TEDE radiation dose to the guideline value is [also large]. * * * In all cases, there is significant margin between the newly calculated post-LOCA TEDE radiation doses and the corresponding regulatory guideline values. In the sense that the margins to the germane regulatory guideline values are still large, the new values of TEDE radiation doses are comparable to the equivalent TEDE associated with the post-LOCA radiation doses currently listed in the UFSAR. Furthermore, these margins for the design basis LOCA do not significantly decrease with insertion of the 4 MOX LFAs. Therefore, the proposed amendment is determined to not result in a significant increase in accident consequences.

AST analyses also were completed for the design basis locked rotor accident (LRA) and rod ejection accident (REA). Again, these design basis accidents were postulated to occur at a Catawba nuclear unit operating with either an all LEU core or with 4 MOX LFAs. The TEDEs following these design basis accidents were compared to the equivalent TEDEs associated with the current license basis analyses. The equivalent TEDEs were computed from the post-accident whole body and thyroid radiation doses using the method prescribed in Regulatory Guide 1.183, as noted above. TEDEs only at offsite locations were compared as post-accident control room radiation doses are not reported for these design basis accidents in the Catawba UFSAR.

* * * * * * *

For the EAB, LPZ, and control room, the post-LRA TEDEs are seen to increase from the values equivalent to the radiation doses from the current license basis analyses. (This is attributed primarily to the increase in assumed fraction of the fuel pins with clad failure following a design basis LRA at Unit 2. * * *) However, the margins to the acceptance criteria of 2.5 Rem at the offsite locations and 5 Rem in the control room are still significant.

* * * * * * *

For the EAB, LPZ, and control room, the post-REA TEDEs are seen to increase from the values equivalent to the radiation doses from the current license basis analyses, as they did for the design basis LRA. (This is attributed to a number of reasons. These include increase in the fraction of gap activity released to containment, inclusion of limiting radial peaking in the source term, and inclusion of alkali metals.) However, the margins to the acceptance criteria of 6.3 Rem at the offsite locations and 5 Rem in the control room are still significant * * *.

The changes proposed to the TS for Containment Penetrations are editorial in nature and will have no effect upon accident consequences.

The changes proposed to the VFTP TS for the AVS, ABFVES, and FHVES will not result in a significant increase in any accident consequences. The changes to make the penetration values for Unit 2 consistent with Unit 1 for the AVS, ABFVES, and FHVES are acceptable because the appropriate safety factors as delineated in the applicable regulatory guideline documents are still maintained. The change to the flowrate specified for the ABFVES is consistent with the design basis operation of this system. Also, the editorial changes proposed to the VFTP TS will have no impact on any accidents.

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

Second Standard

Does operation of the facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated? No.

This proposed amendment does not involve addition, removal, or modification of any plant system, structure, or component. These changes will not affect the operation of any plant system, structure, or components as directed in plant procedures.

The analysis performed in support of this license amendment request, together with the analyses of the design basis fuel handling accident and weir gate drop reported in previously submitted and NRC approved license amendment requests, includes full scope implementation of AST methodology. This analysis does not represent any change in the post-accident operation of any plant system, structure, or component.

Operation of the facility in accordance with this amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Third Standard

Does operation of the facility in accordance with the proposed amendment involve a significant reduction in the margin of safety? No.

Margin of safety is related to confidence in the ability of fission product barriers to perform their design functions following any of their design basis accidents. These barriers include the fuel cladding, the Reactor Coolant System, and the containment. The performance of these barriers either during normal plant operations or following an accident will not be affected by the changes associated with the license amendment request.

The AVS is associated with the containment fission product barrier. Its post-accident operation will not be affected by implementation of the amendment to its TS. The operation of the ABFVES either during normal plant operations or following an accident will not be affected by implementation of the amendment to its TS. The operation of the FHVES either during normal plant operations or following an accident will not be affected by implementation of the amendment to its TS. The operation of the CRAVS either during normal plant operations or following an accident will not be adversely affected by the proposed changes to its TS Bases. The operation of Containment Penetrations following an accident will not be adversely affected by the proposed change to its TS.

As noted, an analysis of radiological consequences of the design basis LOCA at Catawba Nuclear Station has been performed in support of this license amendment request. The design basis LOCA scenarios were selected based on extensive evaluations of Catawba, its design basis, and its anticipated response to a design basis LOCA. Credit was taken only for safety related systems, structures, and components in simulating the mitigation of radiological consequences of the LOCA. Limiting values were taken for performance characteristics of the Class 1E systems modeled in the analysis. The radiological consequences (TEDE radiation doses at the EAB, LPZ, and in the control room) are within the regulatory guideline values with significant margin.

The changes proposed to the VFTP TS for the AVS, ABFVES, and FHVES will not result in a significant reduction in the margin of safety. These changes are supported by regulatory guidance documents, and are consistent with existing system operation. Also, the editorial changes proposed to the VFTP TS will not have any impact on safety.

Operation of the facility in accordance with the proposed amendment does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: John A. Nakoski.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: September 30, 2004, as supplemented by letter dated April 26, 2005.

Description of amendment request: The proposed amendment would change the existing steam generator (SG) tube surveillance program to be consistent with that being proposed by the Technical Specifications Task Force (TSTF) in TSTF–449. These proposed changes would revise Technical Specification (TS) 1.1 on definitions, TS 3.4.13 on reactor coolant system operational leakage, TS 5.5.9 on SG program, and TS 5.6.7 on SG tube inspection reports, and add a new TS 3.4.16 on SG tube integrity. Also, as a result of the licensee replacing the SGs with SGs having a new Alloy 690 thermally treated tubing design, the TSs would be revised to reflect this replacement. The September 30, 2004, application was noticed in the **Federal Register** on November 9, 2004 (69 FR 64987).

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 requires a Steam Generator Program that includes performance criteria that will provide reasonable assurance that the steam generator (SG) tubing will retain integrity over the full range of design basis operating conditions (including startup, power operation, hot standby, cooldown, anticipated transients and postulated accidents). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational LEAKAGE. These criteria assure that the probability of an accident will not be increased.

The primary to secondary accident induced leakage rate for any design basis accidents, other than an SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. [The primary to secondary accident induced leakage rate is relatively inconsequential for the SG tube rupture analysis.] The operational LEAKAGE performance criterion meets current NRC regulations and NEI [Nuclear Energy Institute] 97–06 criteria for reactor coolant system (RCS) operational primary to secondary LEAKAGE through any one SG of 150 gallons per day. These criteria assure that accident doses will stay within regulatory and licensing basis limits.

Therefore, the proposed change does not affect the probability or consequences of any ANO–1 [Arkansas Nuclear One, Unit 1] analyzed accidents.

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 performance based requirements are an improvement over the requirements imposed by the current technical specifications. Implementation of the proposed Steam Generator Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. The proposed 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. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

Steam generator 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 Steam Generator Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the Steam Generator Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current technical specifications.

Therefore, the margin of safety is not changed by the proposed change to the ANO–1 TSs.

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

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

NRC Section Chief: Allen G. Howe.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: March 30, 2005.

Description of amendment request: The proposed amendment adopts the following Nuclear Regulatory Commission (NRC) approved Technical Specification Task Force (TSTF) changes that affect the Boiling Water Reactor (BWR)/6 Improved Standard Technical Specifications:

TSTF No.	Description	TS section affected	Type of change
046, Rev. 1	Clarify the Containment Isolation Valve Surveillance Requirement (SR) to apply only to automatic isolation valves.	SR 3.6.1.3.4 SR 3.6.4.2.2 SR 3.6.5.3.3	Administrative.
222, Rev. 1	Control Rod Scram Time Testing	SR 3.1.4.1 SR 3.1.4.4	Administrative.
264, Rev.	Delete flux monitors specific overlap SRs	SR 3.3.1.1.5 SR 3.3.1.1.6 Table 3.3.1.1–1	Less Restrictive.
275, Rev. 0	Clarify requirements for Diesel Generator (DG) start signal on Reactor Pressure Vessel (RPV) level—low, low, low during RPV cavity flood-up.	Table 3.3.5.1–1, Footnote (a).	Administrative.
276, Rev. 2	Revise DG full load rejection test	SR 3.8.1.9 SR 3.8.1.10 SR 3.8.1.14	Less Restrictive.
300, Rev. 0	Eliminate DG loss of coolant accident-Start SRs while in shutdown when emergency core cooling system is not required.	SR 3.8.2.1	Less Restrictive.
322, Rev. 2	Secondary Containment Integrity SRs	SR 3.6.4.1.3 SR 3.6.4.1.4	Administrative.
400, Rev. 1	Clarification of SR on bypass of DG automatic trips	SR 3.8.1.13	Administrative.
416, Rev. 0	SR 3.5.1.2 Notation	LCO 3.5.1 SR 3.5.1.2 LCO 3.5.2 SR 3.5.2.4	Administrative.

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 changes to the TS [Technical Specifications] involve both administrative and less restrictive changes. The administrative changes involve wording changes that clarify requirements without changing the original intent. As such, these types of changes do not affect initiators of analyzed events and do not affect the mitigation of any accidents or transients.

The less restrictive changes involve modifications to Surveillance Requirements. The modified Surveillance Requirements do not cause the plant to be operated in a new or different manner and the required equipment continues to be tested in a manner and at a frequency necessary to provide confidence that the equipment can perform its intended safety function. Consequently, no initiators to accidents previously evaluated are affected and no mitigating equipment assumed in accidents previously evaluated is adversely affected.

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 changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed), do not change the design function of any equipment, and do not change the methods of normal plant operation. Accordingly, the proposed changes do not create any new credible failure mechanisms, malfunctions, or accident initiators not previously considered in the GGNS [Grand Gulf Nuclear Station] 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 change involve a significant reduction in a margin of safety?

Response: No.

The proposed changes have no affect on any safety analysis assumptions or methods of performing safety analyses. The changes do not adversely affect system OPERABILITY or design requirements and the equipment continues to be tested in a manner and at a frequency necessary to provide confidence that the equipment can perform its intended safety functions. 10 CFR 50.36 (c)(3) requires the TS to include Surveillance Requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. The GGNS TS Surveillance Requirements will continue to provide this assurance with the proposed adoption of the NRC approved TSTF changes.

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

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

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Allen G. Howe.

Entergy Nuclear Operations, Inc., Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: December 14, 2004.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.3.G, ''Scram Discharge Volume [SDV],'' to allow vent or drain lines with inoperable valves to be isolated instead of requiring the valves to be restored to Operable status or to be in Hot Shutdown within 12 hours.

The NRC staff issued a Notice of Opportunity for Comment in the **Federal Register** on February 24, 2003 (68 FR 8637), on possible amendments to revise the action for one or more SDV vent or drain lines with an inoperable valve, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line-item improvement process. The NRC staff subsequently issued a Notice of Availability of the models for referencing license amendment applications in the **Federal Register** on April 15, 2003 (68 FR 18294). The licensee affirmed the applicability of the model NSHC determination (modified slightly as a result of the Pilgrim Nuclear Power Station TS format) in its application dated December 14, 2004.

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

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

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

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

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

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

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

Based on the reasoning presented above, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: J. M. Fulton, Esquire, Assistant General Counsel, Pilgrim Nuclear Power Station, 600 Rocky Hill Road, Plymouth, Massachusetts 02360–5599.

NRC Section Chief: Darrell J. Roberts.

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

Date of amendment request: January 21, 2005.

Description of amendment request: The proposed change permanently revises Isolation Condenser (IC) Technical Specifications (TS) Section 3.5.3, ''IC System.'' Specifically, surveillance requirement SR 3.5.3.4 is modified by the addition of a note which states the IC System heat removal capability surveillance is not required to be performed until 12 hours after adequate reactor power is achieved to perform the test.

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:

According to 10 CFR 50.92, ''Issuance of amendment,'' paragraph (c), a proposed amendment to an operating license involves a no significant hazards consideration if 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;

(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.

In support of this determination, an evaluation of each of the three criteria set forth in 10 CFR 50.92 is provided below regarding the proposed license amendment.

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

Response: No.

The design function of the Isolation Condenser (IC) System is to provide reactor core cooling in the event that the reactor becomes isolated from the turbine and the main condenser by closure of the main steam isolation valves (MSIVs). Although the system is an Engineered Safety Feature System, no credit for IC System operation is taken in the accident analysis. The IC System is designed and installed to provide adequate core cooling, thereby mitigating the consequences of this reactor isolation transient (e. g., inadvertent closure of the MSIVs). This transient has been evaluated in the Updated Final Safety Analysis Report (UFSAR) as an event of moderate frequency. The IC system is designed to operate automatically or manually to perform its design function for reactor pressures greater than 150 psig. Since the IC System is not credited, this TS change does not impact any of the assumptions, inputs, or results of the UFSAR reactor isolation analysis.

The addition of the note to the Technical Specifications surveillance requirement does not alter the IC System design function or the processes and parameters by which the system and its components perform its function. The addition of this note allows the plant to enter an operating mode necessary to allow performance of the heat removal capability surveillance. The purpose of this heat removal capability surveillance is to verify proper flow path and the ability to remove a design heat load. The proposed change does not alter the ability or methods used to verify flow path or heat removal capability. Nor does the change alter the acceptance criteria for satisfactory performance. Therefore, the change does not result in an increase in the consequences of a reactor isolation transient. Additionally, there are no IC System malfunctions or component failures that could initiate a reactor isolation transient. The proposed change does not alter the system or its operation and will not change the IC System's impact on initiating accidents or transients. Therefore, this change, and any associated impacts, will not increase the probability of the occurrence of an accident or transient.

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.

The addition of the note to the Technical Specifications surveillance requirement does not alter the IC System design function or the processes and parameters by which the system and its components perform its function. The existing Technical Specification does not provide any limitations on when the IC System heat removal capability surveillance may be performed. Present plant procedures perform this surveillance at between 60% and 75% reactor power to ensure sufficient steam is available to simulate design heat loads. The addition of the note to the Technical Specification does not create any constraints on plant operating conditions associated with performance of the IC System heat removal capability surveillance. Operation of the IC System to perform the required surveillance in operating Modes 1, 2, or 3 has been previously evaluated and is presently allowed.

The proposed change does not modify the procedural steps for performing the Technical Specification required surveillance. Nor does the change alter the methodology for evaluating acceptable performance. No physical or operational changes are made that could result in plant or system operation in conditions not 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 amendment involve a significant reduction in a margin of safety?

Response: No.

Technical Specification surveillance requirement SR 3.5.3.4 requires verification of the IC System's heat removal capability every 60 months. This surveillance ensures the proper system flow path and ability to remove decay heat following a reactor isolation. The methodology and acceptance criteria for this surveillance are not impacted by this change. Technical Specifications presently allow performance of this surveillance in Modes 1, 2, or 3 and plant procedures presently perform this surveillance in Mode 1. The surveillance is still required to demonstrate the IC System design basis capability of removing the design requirement of 252.5 x 10^6 Btu/hr. Other IC System surveillance requirements are not directly or indirectly impacted by this change. Additionally, this amendment request results in no change to the system's actuation response, operation, or setpoints for performance.

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

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

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

NRC Section Chief: Gene Y. Suh.

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

Date of amendment request: June 11, 2004.

Brief description of amendment request: The proposed license amendment request would relocate surveillance test intervals of various Technical Specification (TS) surveillance requirements to a new program controlled in accordance with the requirements of 10 CFR 50.59. The proposed changes would add a new program, the Surveillance Frequency Control Program, to the Administrative Controls section of the TSs. The proposed amendment is a pilot submittal in support of the Boiling Water Reactor Owners' Group Risk-Informed Initiative 5b, ''Relocate Surveillance Test Intervals to Licensee Control.''

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 involves the relocation of various surveillance test intervals from Technical Specifications (TS) to a licensee-controlled program and is administrative in nature. The proposed change does not involve the modification of any plant equipment or affect basic plant operation. The proposed change will have no impact on any safety related structures, systems or components. Surveillance test intervals are not assumed to be an initiator of any analyzed event, nor are they assumed in the mitigation of consequences of accidents. The surveillance requirements themselves will be maintained in TS[s] along with the applicable Limiting Conditions for Operation (LCOs) and Action statements. The surveillances performed at the intervals specified in the licensee-controlled program will assure that the affected system or component function is maintained, that the facility operation is within the Safety Limits, and that the LCOs are met.

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 previously evaluated?

Response: No. The proposed change does not involve any physical alteration of plant equipment and does not change the method by which any safety-related system performs its function or is tested. As such, no new or different types of equipment will be installed, and the basic operation of installed equipment is unchanged. The methods governing plant operation and testing remain consistent with the safety analysis assumptions.

Therefore, the proposed change will 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 is administrative in nature, does not negate any existing requirement, and does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there are no changes being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. Margins of safety are unaffected by relocation of the surveillance test intervals to a licensee-controlled program.

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

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

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

NRC Section Chief: Darrell J. Roberts.

Exelon Generation Company, LLC, Docket No. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: July 22, 2004, as supplemented December 3, 2004.

Description of amendment request: The proposed amendment would modify the operability and surveillance requirements in Technical Specification 3/4.1.3, ''Control Rods.'' Specifically, the proposed changes would (1) exclude a fully inserted immovable control rod from the shutdown action statement, (2) eliminate consideration of control rod drive water pressure in the action statement, and (3) limit the 24-hour exercise test of other control rods to a one-time occasion following detection of an immovable control rod.

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 first proposed change would exclude fully inserted immovable control rods from consideration in the plant shutdown action statement. An inoperable control rod that has been fully inserted, and disarmed, has satisfied the safety function of that control rod since it is in a position of maximum contribution to shutdown capability. A plant shutdown for this situation would result in an unnecessary plant thermal cycle without any compensatory safety benefit. Under the proposed change, inoperable inserted rods would continue to be counted in the operability requirement precluding power operation with more than 8 inoperable control rods.

The second proposed change removes the control rod drive (CRD) water pressure limits from the insertion capability test of inoperable, non-stuck, control rods. Reactor pressure, assisted by a pre-charged accumulator, provides the driving force for the rapid shutdown of the reactor (scram), independent of the CRD water pressure. Variation of this pressure is not an indicator of a degraded control rod, and does not inhibit the safety function of the control rod. Control rod scram and exercise testing requirements assure the operability of the CRD system. The proposed change would eliminate the need to unnecessarily insert a control rod into the core if it could not be repositioned using the normal drive water pressure setting.

The third proposed change would limit the increased frequency surveillance requirement (every 24 hours) exercise test of withdrawn control rods upon discovery of an immovable control rod to a one-time test in lieu of every 24 hours. A one-time 24-hour test is sufficient to determine if a generic control rod problem exists. Under the proposed change, following the 24-hour test, and in absence of any additional detectable problems, the control rod exercise test would revert back to a normal testing frequency. Repetitive 24-hour tests [have] the potential to reduce the operable lifespan of hydraulic control unit components and increases the potential for a reactivity management event.

The proposed changes will not impede the ability of the surveillance requirements to detect control rod degradation, or inhibit the control rod drive system from performing its designed safety function.

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

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

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

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

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

Response: No. A fully inserted [control] rod has satisfied its safety function by being in the position of maximum contribution to shutdown reactivity. Eliminating the CRD water pressure limits does not impact scram capability. Further, the proposed changes will eliminate extended accelerated control rod testing that may shorten the lifespan of control components without any compromise in the detection of control rod operability problems. The proposed changes would not impact control rod operability and surveillance requirements that are necessary to assure that the control rod system will perform its designed safety function.

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

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

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

NRC Section Chief: Darrell J. Roberts.

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

Date of amendment request: April 20, 2005.

Description of amendment request: The proposed amendment would revise the technical specifications (TSs) to replace plant-specific position titles with generic position titles. The proposed changes are consistent with NUREG–1430, ''Standard Technical Specifications—Babcock and Wilcox Plants,'' Revision 3. Also, the licensee proposes to delete TS 6.7, ''Safety Limit Violation or Protective Limit Violation,'' including a change to TS 2.1.2, ''Safety Limits and Limiting Safety System Settings—Reactor Core,'' associated with the deletion of TS 6.7. Additionally, the licensee proposes to relocate to the Technical Requirements Manual (TRM), the Process Control Program requirements from TS 6.8, ''Procedures and Programs,'' and from TS 6.14, ''Process Control Program (PCP).'' Associated with this change, TS Definition 1.30, ''Process Control Program,'' is proposed to be deleted. Also, TS 6.15, ''Offsite Dose Calculation Manual (ODCM),'' is proposed to be modified to eliminate the requirement that changes to the ODCM be reviewed and accepted by the Plant Operations Review Committee (PORC). Lastly, the licensee proposes to revise in the TS the title, ''Industrial Security Plan'' to ''Physical Security Plan.''

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 changes affect the requirements for the administrative controls section of the Technical Specifications. The proposed changes are primarily intended to make the plant-specific position/organizational titles found in the administrative controls section of the Technical Specifications more generic. The proposed changes do not affect any plant structures, systems, and components, and have no effect on plant operations. The proposed changes are administrative and do not affect any existing limits. Accident initial conditions, probability, and assumptions remain as previously analyzed. The proposed changes will have no effect on accident initiation frequency. The proposed changes do not invalidate the assumptions used in evaluating the radiological consequences of any accident. 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 are administrative and do not introduce any new or different accident initiators. 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 are administrative and will not have a significant effect on any 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 E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Gene Y. Suh.

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

Date of amendment request: April 22, 2005.

Description of amendment request: The proposed amendment would revise the technical specifications (TSs) related to fuel handling and storage. Specifically, the proposed change is to reflect that spent fuel storage racks are no longer installed in the cask pit or transfer pit and that there are no longer any low-density fuel storage racks in the spent fuel pool. Additionally, the proposed changes would relocate the requirements of TS 3/4.9.7, ''Crane Travel—Fuel Handling Building,'' to the Technical Requirements Manual (TRM).

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 amendment would relocate the requirements of TS 3/4.9.7 to the DBNPS [Davis-Besse Nuclear Power Station] TRM. Any subsequent changes to the TRM would require evaluation under the appropriate regulatory processes (e.g., 10 CFR 50.59). The proposed relocation of TS 3/4.9.7 does not affect any accident initiators. The relocated TRM requirements will assure the initial conditions assumed in the analysis of a fuel handling accident are maintained. The proposed change does not affect the ability of plant equipment to mitigate the consequences of any accident. The proposed changes to reflect that fuel storage racks are no longer installed in the cask pit or transfer pit and that low density fuel storage racks are no longer installed in the spent fuel pool are consistent with the current plant configuration. The proposed changes do not affect any accident initiators. The revised requirements will continue to assure the capability to mitigate the consequences of a fuel handling accident in the fuel storage area. 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 relocation of TS 3/4.9.7 to the TRM does not alter the design, operation, or testing of any structure, system, or component. The proposed changes to reflect that fuel storage racks are no longer installed in the cask pit or transfer pit and that low density fuel storage racks are no longer installed in the spent fuel pool are consistent with the current plant configuration. No new accident initiators are created. 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 relocation of TS 3/4.9.7 to the TRM does not alter the design, operation, or testing of any structure, system, or component. The proposed changes to reflect that fuel storage racks are no longer installed in the cask pit or transfer pit and that low density fuel storage racks are no longer installed in the spent fuel pool are consistent with the current plant configuration and do not adversely affect the ability of any structure, system, or component to perform its safety function. 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: Mary E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Gene Y. Suh.

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

Date of amendment request: May 2, 2005.

Description of amendment request: The proposed amendment would revise technical specification (TS) Figure 2.1–1, ''Reactor Core Safety Limit'' and TS Table 2.2–1, ''Reactor Protection System Instrumentation Trip Setpoints.'' These TS revisions would support the use of Framatome Mark B–HTP fuel in the reactor.

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 changes include a revision of the Reactor Core Safety Limits specified in Technical Specification (TS) Section 2.1.1, and a revision of the Reactor Protection System (RPS) Reactor Coolant System (RCS) Pressure-Temperature setpoint Allowable Value provided in TS Section 2.2.1. The proposed changes preserve the design DNB [departure from nucleate boiling] Ratio safety criterion that there shall be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a departure from nucleate boiling during normal operation or events of moderate frequency. Further, there are no evaluated accidents in which the fuel cladding or fuel assembly structural components are assumed to arbitrarily fail as an accident initiator. The fuel handling accident analysis assumes that the cladding does, in fact, fail as a result of an undefined fuel handling event. However, the probability of an accident initiator for the fuel handling accident is independent of the parameters changed in this amendment request. In addition, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not alter any assumptions previously made in the radiological consequence evaluations, or affect mitigation of the radiological consequences of an accident previously evaluated.

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

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

Response: No.

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident scenarios, failure mechanisms or single failures are introduced as a result of the proposed. All systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design function.

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

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

Response: No.

The proposed changes do not involve a significant reduction in a margin of safety because extensive analyses of the primary fission product barriers, conducted in support of the proposed changes, have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance with the regulatory acceptance criteria. As appropriate, all evaluations have been performed using methods that have either been reviewed and approved by the Nuclear Regulatory Commission or that are in compliance with applicable regulatory review guidance and standards.

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

NRC Section Chief: Gene Y. Suh.

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

Date of amendment request: January 10, 2005.

Description of amendment request: The amendment request proposes to revise the surveillance interval associated with Technical Specification Surveillance Requirement 4.6.1.3b from once every 6 months to once every 24 months for verification that only one door in each containment air lock can be opened at a time.

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

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

The proposed amendment will neither effect nor change any design function, or method of performing or controlling design functions, or any analysis that verifies the capability of structures, systems and components (SSCs) to perform their designed function(s). The proposed amendment will have no adverse effect on plant operation or its controlled configuration. As a result, the proposed amendment will not change assumptions, or change, degrade or prevent actions described or assumed in accidents evaluated and described in the Seabrook Station Updated Final Safety Analysis Report (UFSAR). The proposed change extends the surveillance interval from 6 months to 24 months to verify proper functioning of the containment air lock interlocks. The proposed change to the Surveillance Requirement testing interval does not adversely affect performance of the Surveillance Requirement that verifies the functional status of the air lock interlock to prevent both air lock doors to be open simultaneously. Containment integrity is not affected by the proposed amendment. The radiological consequences of an event are unchanged, since the functional status of the air lock interlock is not adversely affected and the air lock doors' ability to withstand the maximum expected post accident containment pressure is not adversely affected by the proposed change. Therefore, the proposed amendment does not adversely affect nuclear safety or continued safe operation of Seabrook Station, or result in an increase in the radiological consequences of any accident described in the Seabrook Station UFSAR.

Therefore, it is concluded that the proposed change does not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

The proposed amendment will neither effect nor change any design function, or method of performing or controlling design functions, or any analysis that verifies the capability of structures, systems and components (SSCs) to perform their designed function(s). The proposed amendment will have no adverse effect on plant operation or its controlled configuration. As a result, the proposed amendment will not change assumptions, or change, degrade or prevent actions described or assumed in accidents evaluated and described in the Seabrook Station UFSAR. There are no changes associated with extending the surveillance interval for the air lock interlock that could potentially introduce new failure modes or accident initiators.

Therefore, it is concluded that 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 extends the surveillance interval from 6 months to 24 months to verify proper functioning of the containment air lock interlock. The containment air lock interlocks are normally not challenged and operating experience has shown these components have an excellent surveillance pass rate. Furthermore, increasing the surveillance interval has no affect on the air lock doors' ability to withstand the maximum expected post accident containment pressure. Containment integrity is not affected by the proposed amendment. The proposed amendment will neither effect nor change any design function, or method of performing or controlling design functions, or any analysis that verifies the capability of structures, systems and components (SSCs) to perform their designed function(s). The functional status of the containment air lock interlocks will continue to be verified.

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

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

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

NRC Section Chief: Darrell J. Roberts.

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

Date of amendment request: March 28, 2005.

Description of amendment request: The proposed amendment would extend the expiration date of Facility Operating License (FOL) NPF–86 for Seabrook Station, Unit No. 1 by approximately 3.4 years. The extension would set the date of expiration of the FOL to occur 40 years from the date of issuance of the full-power operating license. Specifically, the FOL, with a current expiration date of October 17, 2026 would be revised to expire on March 15, 2030. This change would allow the recapture of zero-power and low-power testing time in accordance with SECY–98–296, ''Agency Policy Regarding Licensee Recapture of Low-Power Testing or Shutdown Time for Nuclear Power Plants,'' dated December 21, 1998.

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

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

The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated since it does not involve a change to design configuration or operation of the facility. The proposed change does not effect the source term, containment isolation or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated in the Seabrook Station UFSAR [updated final safety analysis report]. In addition, Seabrook Station Unit [No.] 1 was designed and constructed to ensure a 40-year service life. Design features were incorporated that provide for inspection of structures, systems and components during the 40-year service life. Surveillance, inspection and maintenance practices have been implemented in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and the unit Technical Specifications to provide assurance that any degradation in plant safety-related equipment will be identified and corrected to provide continued safe operation of the unit throughout the duration of the facility operating license.

The recapture period requested by this amendment is for 3.4 years. This time is insignificant from an aging effect perspective when considered in conjunction with the surveillance, inspection and maintenance programs implemented to provide early indication of degradation in plant safety-related equipment. Continual maintenance and testing provides for continued safe operation of the unit throughout the duration of the facility operating license.

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?

The proposed amendment revises the expiration of the facility operating license such that the expiration of the facility operating license is based upon issuance of the FPOL [full-power operating license] and not upon issuance of the ZPOL/LPOL [zero-power operating license/low-power operating license]. The proposed change[s] do[es] not involve physical alteration of plant systems[,] structures or components or changes in parameters governing the manner in which the plant is operated and maintained.

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?

The proposed amendment revises the expiration of the facility operating license such that the expiration of the facility operating license is based upon issuance of the FPOL and not upon issuance of the ZPOL/LPOL. No physical changes are being made to the design features or operation of the facility.

Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel cladding, reactor coolant system pressure boundary and the containment structure) to limit the radiological dose to the public and control room operators in the event of an accident. The proposed amendment to the facility operating license has no impact on the margin of safety and robustness provided in the design and construction of the facility. In addition, the proposed amendment will not relax any of the criteria used to establish safety limits, nor will the proposed amendment relax safety system settings or limiting conditions of operation as defined in the Technical Specifications.

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

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

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

NRC Section Chief: Darrell J. Roberts.

Nuclear Management Company, LLC, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan

Date of amendment request: April 26, 2005.

Description of amendment request: The proposed amendment would revise Technical Specifications (TS) 5.6.5.b., ''Core Operating Limits Report (COLR),'' to add the Palisades-specific fuel assembly growth model to the analytical methods referenced in the TS.

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 license amendment augments an existing analytical method used to determine the core operating limits per Technical Specification 5.6.5.b. Accidents previously evaluated will be unaffected because they will continue to be analyzed using applicable methodologies approved by the Nuclear Regulatory Commission to ensure all required safety limits are met. The proposed amendment does not affect the acceptance criteria for any Final Safety Analysis Report (FSAR) safety analysis analyzed accidents and anticipated operational occurrences. As such, the proposed amendment does not increase the probability or consequences of an accident. The proposed amendment does not involve operation of the required structures, systems or components (SSCs) in a manner or configuration different from those previously recognized or evaluated.

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

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 involve a physical alteration of any SSC or a change in the way any SSC is operated. The proposed amendment does not involve operation of any required SSCs in a manner or configuration different from those previously recognized or evaluated. No new failure mechanisms will be introduced by the changes being requested.

Therefore, the proposed amendment does 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.

The proposed amendment does not, by itself, introduce a failure mechanism. The proposed amendment does not involve any physical changes to the plant or manner in which the plant is operated. The proposed changes do not affect the acceptance criteria for any FSAR safety analysis analyzed accidents or anticipated operational occurrences. All required safety limits would continue to be analyzed using methodologies approved by the Nuclear Regulatory Commission.

Therefore, the proposed amendment would 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: Jonathan Rogoff, Esquire, Vice President, Counsel & Secretary, Nuclear Management Company, LLC, 700 First Street, Hudson, WI 54016.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: November 23, 2004.

Description of amendment request: The proposed amendment revises the descriptive wording of Technical Specifications Table 1–1, ''RPS [reactor protection system] Limiting Safety System Settings,'' for the Reactor Trip setpoint for Low Steam Generator Water Level to relocate unnecessary detail and converts Technical Specifications Section 4.0, Design Features, to be consistent with NUREG–1432, Revision 3, ''Standard Technical Specifications for Combustion Engineering Plants.'' These changes will be needed to support the operation of Fort Calhoun Station (FCS) after major components (steam generators, pressurizer, and reactor vessel head) are replaced in 2006.

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

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

The proposed changes are not related to an initiator of any previously evaluated accident. The proposed changes revise descriptive information only, and will not prevent safety systems from performing their accident mitigation function as assumed in the safety analysis.

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

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

The proposed changes only relocate descriptive information in the Technical Specifications to the USAR [Updated Safety Analysis Report]. Modifications will not be made to existing equipment nor will any new or different types of equipment be installed. The proposed changes to the Technical Specifications will not alter assumptions made in safety analysis and licensing bases.

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

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

The proposed administrative changes only relocate descriptive information in the FCS Technical Specifications to the USAR, and have no effect on safety margins.

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

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

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

NRC Section Chief: Robert A. Gramm.

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: September 8, 2004.

Description of amendment request: The proposed amendments would change the SSES 1 and 2 Technical Specifications (TSs) limiting conditions for operation (LCO) 3.8.4, ''DC Sources-Operating,'' to incorporate the Technical Specifications Change Task Force (TSTF) 16, Revision 2, and other unrelated editorial changes.

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 of occurrence [sic] or consequences of an accident previously evaluated?

Response: No.

The Technical Specification allowed Completion Time for any inoperability is not an initiator to any accident sequence analyzed in the Final Safety Analysis Report (FSAR). The changes do not involve any physical change to structures, systems, or components (SSCs) and does not alter the method of operation or control of SSCs. The current assumptions in the safety analysis regarding accident initiators and mitigation of accidents are unaffected by these changes. No additional failure modes or mechanisms are being introduced and the likelihood of previously analyzed failures remains unchanged.

Operation in accordance with the proposed Technical Specification (TS) ensures that the AC distribution system and supported equipment functions remain capable of performing the function as described in the FSAR. Therefore, the mitigative functions supported by the system will continue to provide the protection assumed by the analysis.

The correction of typographical errors, changes in format and the deletion of a no longer required one-time exemption are administrative changes.

Therefore, this 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 involve a physical alteration of the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There are no setpoints, at which protective or mitigative actions are initiated, affected by this change. This change will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alterations in the procedures that ensure the plant remains within analyzed limits are being proposed, and no changes are being made to the procedures relied upon to respond to an off-normal event as described in the FSAR. The correction of typographical errors, changes in format and the deletion of a no longer required one-time exemption are administrative changes. As such, no new failure modes are being introduced. The change does not alter assumptions made in the safety analysis and licensing basis.

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

Response: No.

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed change is acceptable because the restoration times for deenergized AC distribution subsystems has been previously evaluated in Unit 2 Amendment No. 148. Additional margin of safety is gained with the inclusion of the requirement to enter applicable actions for inoperable Class lE battery chargers as a result of inoperable AC bus(es). The correction of typographical errors, changes in format and the deletion of a no longer required one-time exemption are administrative changes. Therefore the plant response to analyzed events will continue to provide the margin of safety assumed by the analysis.

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 Section Chief: Richard J. Laufer.

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: January 28, 2005.

Description of amendment request: The proposed amendments would change the SSES 1 and 2 Technical Specifications (TSs) 5.5.6, ''Inservice Testing Program,'' to replace the reference to American Society of Mechanical Engineers (ASME) Boiler and PressureVessel Code, Section XI, with a reference to ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) as the source of requirements for the inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. These changes are consistent with the implementation of the SSES 1 and 2 Third 10-Year Interval Inservice Testing Program in accordance with the requirements of 10 CFR 50.55a(f).

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

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

Response: No.

The proposed changes revise Technical Specification 5.5.6 for SSES Units 1 and 2 to conform to the requirements of 10 CFR 50.55a(f) regarding the inservice testing of pumps and valves for the Third 10-Year Interval. The current Technical Specifications reference the ASME Boiler and Pressure Vessel Code, Section XI, requirements for the inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The proposed changes would reference the ASME OM Code, which is consistent with 10 CFR 50.55a(f) and accepted for use by the NRC. The proposed changes are administrative in nature.

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 5.5.6 for SSES Units I and 2 to conform to the requirements of 10 CFR 50.55a(f) regarding the inservice testing of pumps and valves for the Third 10-Year Interval. The current Technical Specifications reference the ASME Boiler and Pressure Vessel Code, Section XI, requirements for the inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The proposed changes would reference the ASME OM Code, which is consistent with 10 CFR 50.55a(f)and accepted for use by the NRC. The proposed changes are administrative in nature.

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 changes revise Technical Specification 5.5.6 for SSES Units I and 2 to conform to the requirements of 10 CFR 50.55a(f) regarding the inservice testing of pumps and valves for the Third 10-Year Interval. The current Technical Specifications reference the ASME Boiler and Pressure Vessel Code, Section XI, requirements for the inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The proposed changes would reference the ASME OM Code, which is consistent with 10 CFR 50.55a(f) and accepted for use by the NRC. The proposed changes are administrative in nature.

Therefore, the proposed change[s] does [sic] 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 Section Chief: Richard J. Laufer.

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: February 7, 2005.

Description of amendment request: The proposed amendments would change the SSES 1 and 2 Technical Specifications (TSs) for ''Secondary Containment,'' limiting condition for operation (LCO) 3.6.4.1, by revising the frequency note applicable to Surveillance Requirements (SR) 3.6.4.1.4 and SR 3.6.4.1.5. The revised note requires each SR be performed with the 3 zone configuration every 60 months.

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 does not involve a significant increase in the probability of an accident previously evaluated because neither Secondary Containment nor the Standby Gas Treatment System is an initiator of an accident. Both mitigate accident consequences.

The consequences of a Design Basis Analysis-Loss of Coolant Accident (DBA–LOCA) have been evaluated in the FSAR [final safety analysis report]. Revising the surveillance frequency to require the most limiting configurations to be tested with the 60-month period rather than just the three zone configuration provides assurance that the most limiting secondary containment configuration is tested every 60 months in accordance with the original intent of the surveillance frequency. The proposed change also provides added assurance of acceptable performance within the analysis assumptions of the FSAR. The radiological evaluation of DBA–LOCA doses, including doses offsite, control room habitability, and exposures for personnel are not impacted.

Therefore, this 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 changes do not involve a physical alteration of the plant. No new or different [kind] of equipment will be installed nor will there be changes in methods governing normal plant operation.

The potential for the loss of plant systems or equipment to mitigate the effects of an accident is not altered.

The proposed changes do not require any new operator response or introduce any new opportunities for operator error not previously considered.

Thus, this 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 does not involve a significant reduction in [a] margin of safety.

The surveillance test change ensures all the secondary containment configurations are tested within a 60-month period when only one configuration was previously required to be tested. This change has a positive effect on the margin of safety as it provides more restrictive testing requirement that will provide added assurance of acceptable secondary containment performance.

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 Section Chief: Richard J. Laufer.

PPL Susquehanna, LLC, Docket No. 50–388, Susquehanna Steam Electric Station, Unit 2 (SSES 2), Luzerne County, Pennsylvania

Date of amendment request: March 18, 2005.

Description of amendment request: The proposed amendment would revise the SSES 2 Technical Specification (TS) 3.3.8.1, ''Loss of Power (LOP) Instrumentation,'' to: (1) clarify that Condition A applies to inoperable instrumentation other than during the performance of Surveillance Requirement (SR) 3.8.1.19 loss-of-coolant accident/loss of offsite power testing on Unit 1 and to revise TS Bases section to clarify that this condition is applicable to both Unit 1 and Unit 2 LOP Instrumentation, (2) add new Condition B to allow the LOP instrumentation for two Unit 1 4.16kV Engineered Safeguards System buses in the same Division to be inoperable for up to 8 hours for the performance of SR 3.8.1.19 on Unit 1. In addition, the proposed amendment would revise the SSES 2 TS 3.8.7, ''Distribution Systems-Operating,'' to: (1) eliminate ''or more'' and the plural to subsystems such that the condition would read ''One Unit 1 AC [alternating current] electrical power distribution subsystem inoperable,'' (2) add new Condition D for two Unit 1 AC electrical power distribution subsystems inoperable.

This will impose an 8-hour Completion Time for restoration of at least one of the two Unit 1 AC distribution subsystems.

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 of occurrence or consequences of an accident previously evaluated?

Response: No.

The Technical Specification allowed Completion Time for any inoperability is not an initiator to any accident sequence analyzed in the Final Safety Analysis Report (FSAR). The changes do not involve any physical change to structures, systems, or components (SSCs) and does not alter the method of operation or control of SSCs. The current assumptions in the safety analysis regarding accident initiators and mitigation of accidents are unaffected by these changes. No additional failure modes or mechanisms are being introduced and the likelihood of previously analyzed failures remains unchanged.

Operation in accordance with the proposed Technical Specification (TS) ensures that the AC distribution system and supported equipment functions remain capable of performing the function as described in the FSAR [final safety analysis report]. Therefore, the mitigative functions supported by the system will continue to provide the protection assumed by the analysis.

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

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

Response: No.

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed change is acceptable because the restoration time for deenergized AC distribution subsystems has been previously evaluated in Unit 2 Amendment No. 148. Therefore[,] the plant response to analyzed events will continue to provide the margin of safety assumed by the analysis.

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 Section Chief: Richard J. Laufer.

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

Date of amendment request: March 4, 2005.

Description of amendment request: The proposed amendment would change Technical Specification (TS) 3.5.1, ''Accumulators,'' to extend the completion time (CT) for Action (a) from 1 hour to 24 hours. The accumulators are part of the emergency core cooling system and consist of tanks partially filled with borated water and pressurized with nitrogen gas. The contents of the tank are discharged to the reactor coolant system (RCS) if, as during a loss-of-coolant accident (LOCA), the coolant pressure decreases to below the accumulator pressure. Action (a) of TS 3.5.1 specifies a CT to restore an accumulator to operable status when it has been declared inoperable for a reason other than the boron concentration of the water in the accumulator not being within the required range. This change was proposed by the Westinghouse Owners Group participants in the TS Task Force (TSTF) and is designated TSTF–370. TSTF–370 is supported by NRC-approved Topical Report WCAP–15049–A, ''Risk-Informed Evaluation of an Extension to Accumulator Completion Times,'' submitted on May 18, 1999. The NRC staff issued a Notice of Opportunity for Comment in the **Federal Register** on July 15, 2002 (67 FR 46542), on possible amendments concerning TSTF–370, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a Notice of Availability of the models for referencing license amendment applications in the **Federal Register** on March 12, 2003 (68 FR 11880). The licensee affirmed the applicability of the following NSHC determination in its application dated March 4, 2005.

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

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

The basis for the accumulator limiting condition for operation (LCO), as discussed in Bases Section 3.5.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of the WCAP–15049, ''Risk-Informed Evaluation of an Extension to Accumulator Completion Times,'' evaluation, the proposed change will allow plant operation with an inoperable accumulator for up to 24 hours, instead of 1 hour, before being required to begin shutdown. The impact of the increase in the accumulator CT on core damage frequency for all the cases evaluated in WCAP–15049 is within the acceptance limit of 1.0E–06/yr for a total plant core damage frequency less than 1.0E–03/yr. The incremental conditional core damage probabilities calculated in WCAP–15049 for the accumulator CT increase meet the criterion of 5E–07 in Regulatory Guides (RGs) 1.174 [''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis''] and 1.177 [''An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications''] for all cases except those that are based on design basis success criteria. As indicated in WCAP–15049, design basis accumulator success criteria are not considered necessary to mitigate large-break LOCA events, and were only included in the WCAP–15049 evaluation as a worst-case data point. In addition, WCAP–15049 states that the NRC has indicated that an incremental conditional core damage frequency greater than 5E–07 does not necessarily mean the change is unacceptable.

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

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

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

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

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

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

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

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

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

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

Based upon the reasoning presented above and the previous discussion of the amendment request, the NRC staff proposes to determine that the requested change does not involve a significant hazards consideration.

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

NRC Section Chief: Darrell J. Roberts.

TXU Generation Company LP, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Units 1 and 2, Somervell County, Texas

Date of amendment request: August 5, 2004, as superceded in its entirety by letter dated March 15, 2005.

Brief description of amendments: The proposed amendments would revise Technical Specification (TS) 3.7.10 entitled ''Control Room Emergency Filtration/Pressurization System (CREFS)'' to extend the Completion Time for ACTION B., ''Two CREFS Trains inoperable due to inoperable Control Room boundary in MODES 1, 2, 3, and 4'' from 24 hours to 14 days for implementation of the Turbine Generator Protection System Digital Modification currently scheduled during the eleventh refueling outage for Unit 1 (1RF11) and the ninth refueling outage for Unit 2 (2RF09). The description of CONDITION E would also be revised for implementation of this modification.

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

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

Response: No.

This is a revision to the Technical Specifications for the CREFS which is a mitigation system designed to minimize in leakage and to filter the Control Room atmosphere to protect the operator following accidents previously analyzed. An important part of the system is the Control Room boundary. The Control Room boundary integrity is not an initiator or precursor to any accident previously evaluated. Therefore, the probability of any accident previously evaluated is not increased. The analysis of the consequences of analyzed accident scenarios under the Control Room breach conditions along with the compensatory actions for restoration of Control Room integrity demonstrate that the consequences of any accident previously evaluated are not increased. Therefore, it is concluded that this change does not significantly increase the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change will not impact the accident analysis. The change will not alter the requirements of the CREFS or its function during accident conditions. The administrative controls and compensatory actions will ensure the CREFS will perform its safety function. No new or different accidents result from the revised Completion Time or the restated TS Condition E. The change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The change does not alter assumptions made in the safety analysis. The proposed change is consistent with the safety analysis assumptions and current plant operating practice. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by these changes. The proposed change will not result in plant operation in a configuration outside the design basis for an unacceptable period or time without compensatory actions and administrative controls. The proposed change does not affect systems that respond to safely shutdown the plant and to maintain the plant in a safe shutdown condition. Therefore the proposed change does not involve a reduction in a margin of safety.

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

Attorney for licensee: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Allen G. Howe.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: September 15, 2004.

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

Date of issuance: April 28, 2005.

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

Amendment No.: 164.

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

Date of initial notice in **Federal Register:** February 01, 2005 (70 FR 5235). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 28, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 1, 2004.

Brief description of amendments: The amendments eliminate the requirements to submit monthly operating reports and occupational radiation exposure reports.

Date of issuance: May 9, 2005.

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

Amendment Nos.: 272 and 249.

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

Date of initial notice in **Federal Register:** February 1, 2005 (70 FR 5236).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 6, 2004.

Brief description of amendment: The amendment deleted the requirements to submit monthly operating reports and occupational radiation exposure reports.

Date of issuance: April 28, 2005.

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

Amendment No.: 166.

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

*Date of initial notice in **Federal Register***: February 1, 2005 (70 FR 5236).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 28, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 19, 2004, as supplemented by letters dated March 8 and March 22, 2005.

Brief description of amendments: The amendments revised the Technical Specifications (TS) 3.8.4, ''DC Sources—Operating'' and TS 3.8.6, ''Battery Cell Parameters'' to allow for the replacement of the existing nickel-cadmium diesel generator batteries with conventional lead-acid batteries.

Date of issuance: April 27, 2005.

Effective date: As of the date of issuance and shall be implemented within 30 days from the date of issuance April 27, 2005.

Amendment Nos.: 223 and 218.

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

Date of initial notice in Federal Register: December 21, 2004 (69 FR 76488). The supplements dated March 8 and March 22, 2005, provided additional information that clarified the application, did not expand the scope of the July 19, 2004, application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 27, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 9, 2004, and as supplemented by letter dated April 1, 2005.

Brief description of amendment: This amendment revises Technical Specifications (TS) Limiting Condition for Operation (LCO) 3.4.11, ''RCS [Reactor Coolant System] Pressure and Temperature (P/T) Limits,'' to replace the P/T curves for inservice leak and hydrostatic testing, non-nuclear heating and cooldown, and nuclear heating and cooldown currently illustrated in TS Figures 3.4.11–1, 3.4.11–2, and 3.4.11–3, respectively.

Date of issuance: May 12, 2005.

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

Amendment No.: 193.

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

*Date of initial notice in **Federal Register:*** August 31, 2004 (69 FR 53102).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 12, 2005.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: December 17, 2004.

Brief description of amendment: The amendment deletes Technical Specification (TS) 5.6.1, ''Occupational Radiation Exposure Report,'' and TS 5.6.4, ''Monthly Operating Reports.''

Date of issuance: May 3, 2005.

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

Amendment No: 167.

Facility Operating License No. NPF–29: The amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 1, 2005 (70 FR 9992).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 3, 2005.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: April 14, 2004, as supplemented on December 15, 2004.

Brief description of amendment: This amendment eliminates secondary containment operability requirements when handling sufficiently decayed irradiated fuel or performing core alterations. The secondary containment is still required to be operable during operations with the potential to drain the reactor vessel.

Date of issuance: April 28, 2005.

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

Amendment No.: 215.

Facility Operating License No. DPR–35: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 12, 2004 (69 FR 60679). The December 15, 2004, 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 April 28, 2005.

No significant hazards consideration comments received: No.

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

Date of amendment request: July 17, 2004, as supplemented by letters dated October 18, 2004, February 2, February 21, March 8, and April 5, 2005.

Brief description of amendment: The amendment revised Technical Specification (TS) 5.3.1, to allow the use of a limited number of lead test assemblies, the use of ZIRLO™ as an acceptable fuel cladding, and to allow a limited substitution of zirconium alloy or stainless steel filler rods for fuel rods, while relocating the maximum fuel enrichment from TS 5.3.1 to TS 5.6.1. TS 6.9.1.11.1 is revised to allow the use of the Westinghouse Nuclear Physics code package and to incorporate the methodology used to support ZIRLO™ cladding material. Additionally, the amendment approved the administrative changes of correcting a referencing report error of the CESEC code and deleting the TS Index from the TSs.

Date of issuance: May 9, 2005.

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

Amendment No.: 200.

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

*Date of initial notice in **Federal Register***: July 20, 2004 (69 FR 43460). The supplements dated October 18, 2004, February 2, February 21, March 8, and April 5, 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 May 9, 2005.

No significant hazards consideration comments received: No.

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

Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Docket Nos. 50–010, 50–237 and 50–249, Dresden Nuclear Power Station, Units 1, 2 and 3, Grundy County, Illinois

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

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

Docket Nos. 50–295 and 50–304, Zion Nuclear Power Station, Units 1 and 2, Lake County, Illinois

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

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

Date of issuance: April 29, 2005.

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

Amendment Nos.: Byron Station, Unit 1—142, Unit 2—142; Braidwood Station, Unit 1—136, Unit 2—136; Dresden Nuclear Power Station, Unit 1—42, Unit 2—214, Unit 3—206; LaSalle County Station, Unit 1—173, Unit 2—159; Quad Cities Nuclear Power Station, Unit 1—225, Unit 2—220; Zion Nuclear Power Station, Unit 1—184, Unit 2—171.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72, NPF–77, DPR–2, DPR–19, DPR–25, NPF–11, NPF–18, DPR–29 and DPR–30: The amendments revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. Date of initial notice in **Federal Register:** April 08, 2005 (70 FR 18061). The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 29, 2005.

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

NRC Section Chief: Gene Y. Suh.

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

Date of application for amendment: October 21, 2004.

Brief description of amendment: The amendments deleted the Technical Specifications (TSs) 6.9.1.5.a and 6.9.1.6 requirements to submit monthly operating reports and annual occupational radiation exposure reports. The change is consistent with Revision 1 of the U.S. Nuclear Regulatory Commission's Technical Specifications Task Force (TSTF) Change Traveler, TSTF–369, ''Elimination of Requirements for Monthly Operating Reports and Occupational Radiation Exposure Reports.''

Date of issuance: April 29, 2005.

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

Amendment Nos.: 175 and 137.

Facility Operating License Nos. NPF–39 and NPF–85. The amendments revised the TSs.

*Date of initial notice in **Federal Register***: June 23, 2004 (69 FR 35067). This TS improvement was announced in the **Federal Register** as part of the Consolidated Line Item Improvement Process. A notice for these TS changes was announced on April 8, 2005 (70 FR 18059). The April 8, 2005, notice incorrectly referenced a January 4, 2005, supplement to the application. This supplement was reference by error. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 29, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 10, 2004.

Brief description of amendment: This amendment deletes the Technical Specifications associated with hydrogen recombiners and hydrogen monitors.

Date of issuance: April 19, 2005.

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

Amendment No.: 135.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register***: February 15, 2005 (70 FR 7767). Add the following statement, if appropriate.

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

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: January 28, 2004, as supplemented by letter dated November, 22, 2004.

Brief description of amendment: This amendment revised technical specifications (TSs) 1.4, ''Frequency,'' 5.5.2, ''Primary Coolant Sources Outside Containment,'' and 5.5.11, ''Safety Function Determination Program,'' by adopting three industry-proposed Standard Technical Specifications (STS) changes, which the Nuclear Regulatory Commission (NRC) has approved and included in Revision 3 of the STSs. These changes are Technical Specifications Task Force (TSTF) traveler numbers 273, 284, and 299. The licensee's request to revise TS 3.3.1.1, ''Reactor Protection System Instrumentation,'' which is associated with TSTF–264 is addressed by the NRC staff by a separate Safety Evaluation.

Date of issuance: May 12, 2005.

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

Amendment No.: 258.

Facility Operating License No. DPR–49: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 13, 2004 (69 FR 19571).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 12, 2005.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of application for amendments: February 10, 2004.

Brief description of amendments: The amendments (1) extended from 1 hour to 24 hours the completion time (CT) for Condition C of technical specification (TS) 3.5.1, which defines requirements for the safety injection accumulators. Condition C of TS 3.5.1 specifies a CT to restore an accumulator to operable status when it has been declared inoperable for a reason other than the boron concentration of the water in the accumulator not being within the required range; (2) deleted Condition B which permits one or both accumulators to be inoperable, by removing power to the accumulator isolation valve(s), for maintenance or testing; (3) modified Condition E to remove reference to Condition B; and (4) re-lettered the Conditions and Actions to reflect deletion of Condition B.

Date of issuance: April 28, 2005.

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment Nos.: 217, 222.

Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 13, 2004 (69 FR 19573).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 28, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: May 3, 2004, as supplemented by letters dated February 4, and March 28, 2005.

Brief description of amendments: The amendments revise the licensing to define a new hydraulic analysis methodology for demonstrating functionality of the cooling water (CL) system following a design-basis seismic event. The seismic analysis methodology for the CL system is revised to include (1) evaluation of CL system performance following a seismic event assuming a rupture of a non-seismic pipe at the worst case location, and (2) application of acceptance criteria from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section lll, to demonstrate that the CL system non-seismic piping will maintain pressure boundary integrity with design-basis seismic loads.

Date of issuance: May 10, 2005.

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

Amendment Nos.: 169, 159.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revised the Updated Safety Analysis Report.

Date of initial notice in ***Federal Register:*** July 6, 2004 (69 FR 40677).

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

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 10, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 6, 2004, as supplemented March 14, 2005.

Brief description of amendment: This amendment deletes the Technical Specification requirements associated with hydrogen recombiners and hydrogen monitors.

Date of issuance: May 5, 2005.

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

Amendment No.: 90.

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

Date of initial notice in ***Federal Register:*** February 15, 2005 (70 FR 7768). The supplement dated March 14, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 5, 2005.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 27, 2004.

Brief description of amendments: The amendments delete TS 5.7.1.1.a, ''Occupational Radiation Exposure Report'' and TS 5.7.1.4, ''Monthly Operating Reports.''

Date of issuance: May 10, 2005.

Effective date: May 10, 2005, to be implemented within 60 days of issuance.

Amendment Nos.: Unit 2—195; Unit 3—186.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 1, 2005 (70 FR 5248). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 10, 2005.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–328, Sequoyah Nuclear Plant, Unit 2, Hamilton County, Tennessee

Date of application for amendment: December 2, 2004, as supplemented by letters dated February 15, March 9, and April 11, 2005.

Brief description of amendment: The amendment revises portions of the Sequoyah Unit 2 Technical Specification Surveillance Requirement 4.4.5 to eliminate the requirement to inspect a portion of the tube within the tubesheet region. This will allow any flaws in the region, which is no longer inspected, to remain in service.

Date of issuance: May 3, 2005.

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment No.: 291.

Facility Operating License No. DPR–79: Amendment revises the technical specifications.

Date of initial notice in Federal Register: January 18, 2005 (70 FR 2899). The supplemental letters 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 May 3, 2005.

No significant hazards consideration comments received: No.

TXU Generation Company LP, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: October 28, 2004.

Brief description of amendments: This amendment deletes the Technical Specifications associated with hydrogen recombiners and hydrogen monitors.

Date of issuance: April 21, 2005.

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

Amendment Nos.: 117/117.

Facility Operating License Nos. NPF–87 and NPF–89: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: February 15, 2005 (70 FR 7770).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 21, 2005.

No significant hazards consideration comments received: No.

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

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

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

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

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

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

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

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

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

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

Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: April 18, 2005, as supplemented by letter dated April 19, 2005.

Description of amendment request: The amendment revises Technical Specification (TS) 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' to add changes to the SG inspection scope for Wolf Creek Generating Station for only the current refueling outage 14 and the subsequent operating cycle. Specifically, the amendment modifies the inspection requirements for portions of the SG tubes within the hot leg tubesheet region of the SGs.

Date of issuance: April 28, 2005.

Effective date: Effective the date of issuance, and shall be implemented before entry into Mode 4 in the restart from the current Refueling Outage 14.

Amendment No.: 162.

Facility Operating License No. NPF–42: Amendment revises the technical specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. The Coffey County Republican on April 22 and 26, 2005, and the Emporia Gazette on April 25 and 26, 2005. The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. Comments have been received. The resolution of the comments, the Commission's related evaluation of the amendment, finding of exigent circumstances, state consultation, and final NSHC

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

determination are contained in a safety evaluation dated April 28, 2005.

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Robert A. Gramm.

Dated at Rockville, Maryland, this 16th day of May, 2005.

For the Nuclear Regulatory Commission.

James E. Lyons,

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

[FR Doc. 05–10063 Filed 5–23–05; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

Proposed Collection; Comment Request for Review of an Information Collection: RI 25–49

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: In accordance with the Paperwork Reduction Act of 1995 (Pub. L. 104–13, May 22, 1995), this notice announces that the Office of Personnel Management (OPM) intends to submit to the Office of Management and Budget a request for review of an information collection. RI 25–49, Verification of Full-Time School Attendance, is used to verify that adult student annuitants are entitled to payments. OPM must confirm that a full-time enrollment has been maintained.

Comments are particularly invited on: whether this collection of information is necessary for the proper performance of functions of OPM, and whether it will have practical utility; whether our estimate of the public burden of this collection is accurate, and based on valid assumptions and methodology; and ways in which we can minimize the burden of the collection of information on those who are to respond, through use of the appropriate technological collection techniques or other forms of information technology.

Approximately 10,000 RI 38–45 forms are completed annually. Each form requires approximately 60 minutes to complete. The annual estimated burden is 10,000 hours.

For copies of this proposal, contact Mary Beth Smith-Toomey on (202) 606–8358, FAX (202) 418–3251 or via email to mbtoomey@opm.gov. Please include a mailing address with your request.

DATES: Comments on this proposal should be received within 60 calendar days from the date of this publication.

ADDRESSES: Send or deliver comments to—Pamela S. Israel, Chief, Operations Support Group, Retirement Services Program, Center for Retirement and Insurance Services, U.S. Office of Personnel Management, 1900 E Street, NW, Room 3349, Washington, DC 20415.

FOR FURTHER INFORMATION CONTACT: Cyrus S. Benson, Team Leader, Publications Team, RIS Support Services/Support Group, (202) 606–0623.

U.S. Office of Personnel Management.

Dan G. Blair,

Acting Director.

[FR Doc. 05–10269 Filed 5–23–05; 8:45 am]

BILLING CODE 6325–38–P

OFFICE OF PERSONNEL MANAGEMENT

Proposed Collection; Comment Request for Review of a Revised Information Collection: RI 25–7

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: In accordance with the Paperwork Reduction Act of 1995 (Pub. L. 104–13, May 22, 1995), this notice announces that the Office of Personnel Management (OPM) intends to submit to the Office of Management and Budget a request for review of a revised information collection. RI 25–7, Marital Status Certification Survey, is used to determine whether widows, widowers, and former spouses receiving survivor annuities from OPM have remarried before reaching age 55 and, thus, are no longer eligible for benefits from OPM.

Comments are particularly invited on: Whether this collection of information is necessary for the proper performance of functions of the Office of Personnel Management, and whether it will have practical utility; whether our estimate of the public burden of this collection of information is accurate, and based on valid assumptions and methodology; and ways in which we can minimize the burden of the collection of information on those who are to respond, through the use of appropriate technological techniques or other forms of information technology.

Approximately 2,500 forms are completed annually. Each form takes approximately 15 minutes to complete. The annual estimated burden is 625 hours.

For copies of this proposal, contact Mary Beth Smith-Toomey on (202) 606–8358, FAX (202) 418–3251 or via e-mail to *mbtoomey@opm.gov.* Please include a mailing address with your request.

DATES: Comments on this proposal should be received within 60 calendar days from the date of this publication.

ADDRESSES: Send or deliver comments to—Pamela S. Israel, Chief, Operations Support Group, Retirement Services Programs, U.S. Office of Personnel Management, 1900 E Street, NW., Room 3349, Washington, DC 20415.

FOR FURTHER INFORMATION CONTACT: Cyrus S. Benson, Team Leader, Publications Team, RIS Support Services/Support Group, (202) 606–0623.

U.S. Office of Personnel Management.

Dan G. Blair,

Acting Director.

[FR Doc. 05–10270 Filed 5–23–05; 8:45 am]

BILLING CODE 6325–38–P

OFFICE OF PERSONNEL MANAGEMENT

Proposed Collection; Comment Request for Review of a Revised Information Collection: SF 3102

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: In accordance with the Paperwork Reduction Act of 1995 (Public Law 104–13, May 22, 1995), this notice announces that the Office of Personnel Management (OPM) intends to submit to the Office of Management and Budget a request for review of a revised information collection. SF 3102, Designation of Beneficiary (FERS), is used by an employee or an annuitant covered by the Federal Employees Retirement System to designate a beneficiary to receive any lump sum due in the event of his/her death.

Comments are particularly invited on: whether this collection of information is necessary for the proper performance of functions of OPM, and whether it will have practical utility; whether our estimate of the public burden of this collection is accurate, and based on valid assumptions and methodology; and ways in which we can minimize the burden of the collection of information on those who are to respond, through use of the appropriate technological collection techniques or other forms of information technology.

Approximately 2,037 SF 3102 forms are completed annually. Each form takes approximately 15 minutes to complete. The annual estimated burden is 509 hours.

For copies of this proposal, contact Mary Beth Smith-Toomey on (202) 606–