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 requestors/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/requestor 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. 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.

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, 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.

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(c)(1)(i)–(viii).

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

For further details with respect to this action, see the application for amendment dated February 8, 2007, which is available for public inspection at the Commission's PDR, located at One White Flint North, File Public Area O1 F21, 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.* Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov.*

Dated at Rockville, Maryland, this 7th day of March 2007.

For the Nuclear Regulatory Commission.

Stewart N. Bailey,

Senior Project Manager, Plant Licensing Branch II–2, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation.

[FR Doc. E7–4517 Filed 3–12–07; 8:45 am]

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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 February 15, 2007 through March 1, 2007. The last biweekly notice was published on February 27, 2007 (72 FR 8800).

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, Rulemaking, Directives and Editing 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.*

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 amendment request: February 1, 2007.

Description of amendment request: The proposed license amendment would revise Surveillance Requirement (SR) 3.5.2.8 in Technical Specification 3.5.2, ''ECCS [Emergency Core Cooling System]—Operating,'' to reflect the replacement of the containment recirculation sump suction inlet trash racks and screens with strainers, in response to Nuclear Regulatory Commission (NRC) Generic Letter 2004–02, ''Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors.'' The proposed license amendment would replace ''trash racks and screens'' with ''strainers'' in SR 3.5.2.8.

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 consequences of accidents evaluated in the Updated Final Safety Analysis Report [UFSAR] that could be affected by the proposed change are those involving the pressurization of Containment and associated flooding of the Containment and recirculation of this fluid within the Emergency Core Cooling System (ECCS) or the Containment Spray System (CSS) (*e.g.*, loss-of-coolant accidents [LOCAs]). The proposed change does not impact the initiation or probability of occurrence of any accident. Although the configurations of the existing containment recirculation sump trash racks and screen and the replacement sump strainer cassettes are different, they serve the same fundamental purpose of passively removing debris from the sump's suction supply of the supported system pumps. Removal of trash racks does not impact the adequacy of the pump net positive suction head assumed in the safety analysis. Likewise, the change does not reduce the reliability of any supported systems or introduce any new system interactions. The greatly increased surface area of the new strainer is designed to reduce head loss and reduce the approach velocity at the strainer face significantly, decreasing the risk of impact from large debris entrained in the sump flow stream.

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 containment recirculation sump strainers are a passive system used for accident mitigation. As such, they cannot be accident initiators. Therefore, there is no possibility that this change could create any new or different kind of accident. No new accident scenarios, transient precursors, or limiting single failures are introduced as a result of the proposed change. There will be no adverse effect or challenges imposed on any safety-related system as a result of the change. Therefore, the possibility of a new or different [kind] of accident is not created.

There are no changes which would cause the malfunction of safety-related equipment, assumed to be OPERABLE in the accident analyses, as a result of the proposed Technical Specification change. No new equipment performance burdens are imposed. The possibility of a malfunction of safety-related equipment with a different result is not 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?

The proposed change does not affect the acceptance criteria for any analyzed event nor is there a change to any safety analysis limit. There will be no effect on the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. The proposed change does not adversely affect the fuel, fuel cladding, Reactor Coolant System, or containment integrity. The radiological dose consequence acceptance criteria listed in the Updated Final Safety Analysis Report will continue to be met.

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

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

Attorney for licensee: Carey Fleming, Esquire, Senior Counsel—Nuclear Generation, Constellation Generation Group, LLC, 750 East Pratt Street, 17th floor, Baltimore, MD 21202.

NRC Acting Branch Chief: John P. Boska.

Carolina Power & Light Company, Docket Nos. 50–325 and 5–324 Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of amendments request: December 21, 2006.

Description of amendment request: The proposed amendment would modify technical specification (TS) requirements of TS 3.4.1, ''Recirculation Loops Operating,'' to require the recirculation loops be operated with matched flows versus recirculation pump speeds as currently required. This change affects the Limiting Condition for Operation (LCO) requirements and Surveillance Requirements (SRs) of TS 3.4.1.

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

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

Response: No.

The proposed amendment implements more conservative requirements associated with recirculation loop operation. Specifically, the LCO requirements of TS 3.4.1 and SR 3.4.1.1 are being revised to directly monitor recirculation loop jet pump flows versus recirculation pump speed, eliminating potential non-conservatism associated with relating recirculation loop jet pump flow to recirculation pump speed. These requirements assure that the mismatch between recirculation loop jet pump flows are bounded by the existing design bases analyses. As a result, the proposed change ensures that the consequences of a design bases LOCA [loss-of-coolant accident] remain within the existing evaluation.

The proposed change does not involve a physical change to the Reactor Recirculation system, nor does it alter the assumptions of the accident analyses. Therefore the probability of an accident previously evaluated is not affected.

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

2. Does the proposed 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 change to the Reactor Recirculation system, nor does it alter the assumptions of the accident analyses.

The implementation of more conservative requirements associated with recirculation loop operation does not introduce any new failure modes. 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 amendment implements more conservative requirements associated with recirculation loop operation. These requirements ensure that the Reactor Recirculation system is operated consistent with the initial conditions of the existing design bases analyses. Since the design bases analyses assumptions are unchanged, the proposed change does not involve a reduction in a margin of safety.

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

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

NRC Branch Chief: L. Raghavan.

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

Date of amendment request: December 15, 2006.

Description of amendment request: The amendment would incorporate changes to the Technical Specifications (TS) associated with previously approved industry initiatives. The first change would relocate the Safety Limit Violation specifications from the administrative controls TS section to the safety limit TS sections as approved by TSTF–05–A, ''Deletion of Safety Limit Violation Requirements.'' The second change would incorporate generic position titles, as approved by TSTF–65–A, ''Use of Generic Titles for Utility Positions,'' and incorporates changes approved by NRC Administrative Letter (AL) 95–06, ''Relocation of Technical Specification Administrative Controls Related to Quality Assurance.''

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?

No. The proposed amendment consists of changes to and relocation of administrative TS requirements that were previously generically approved by the NRC. The proposed amendment would not change any of the previously evaluated accidents in the updated safety analysis report (USAR). The administrative controls that are affected by the proposed amendment do not have any function related to preventing or mitigating any of these previously evaluated accidents. The proposed amendment does not affect any systems, structures, or components (SSCs) that have the function of preventing or mitigating any of these previously evaluated accidents. The proposed amendment does not increase the likelihood of the malfunction of an SSC, thus the potential impact on analyzed accidents need not be considered.

Because the proposed amendment is a relocation of administrative requirements that are not associated with preventing or mitigating the consequences of any previously evaluated accidents, there is no affect on the probability or consequences of an accident previously evaluated.

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

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

No. The proposed amendment consists of changes to and relocation of administrative TS requirements previously generically approved by the NRC. This amendment will not change the design function of any SSC or the manner that any SSC is operated. Because this amendment does not change the design function or operation of any SSC, the amendment would not create the possibility of a new or different kind of accident due to credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases.

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?

No. The proposed amendment consists of changes to and relocation of administrative TS requirements previously generically approved by the NRC. The amendment does not alter any design basis safety limit and no safety margins are affected.

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: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Acting Branch Chief: P. Milano.

Duke Power Company LLC, et al., Docket No. 50–413, Catawba Nuclear Station, Unit 1 (Catawba), York County, South Carolina

Date of amendment request: November 22, 2006.

Description of amendment request: The amendment would revise the Catawba Unit 1 Facility Operating License (FOL) to provide for an extension of the time limit to complete the required modification to the Emergency Core Cooling System (ECCS) sump.

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 license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed license amendment delineates a new Unit 1 FOL condition to implement a completion date associated with the ECCS sump strainer modification. The proposed license amendment is administrative in nature and is being submitted to fulfill a commitment made in previous Duke licensing correspondence. Therefore, the proposed license amendment has no effect upon either the probability or consequences of an accident previously evaluated.

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

As stated above, the proposed license amendment is administrative in nature and does not change the manner in which Unit 1 is designed or operated. Therefore, the proposed license amendment cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their intended functions. These barriers include the fuel cladding, the reactor coolant system, and the containment. The performance of these barriers will not be affected by the addition of the proposed FOL condition. Being administrative in nature, the proposed license amendment therefore does not involve a significant reduction in any safety margin.

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, Associate General Counsel and Managing Attorney, Duke Energy Carolinas, LLC, 526 South Church Street, EC07H, Charlotte, NC 28202.

NRC Branch Chief: Evangelos C. Marinos.

Duke Power Company LLC, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: April 11, 2006.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TSs) related to the organizational description in TS 5.2.1

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided it's 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.

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

The proposed change revises an organizational description in TS 5.2.1 to reflect the change of the title of the Vice President Nuclear Generation. The change is solely administrative in nature and has no impact on any accident probabilities or consequences. The change does not affect structures or components in the plant. The change has no affect on any accident previously evaluated. Therefore the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

There are no new accident causal mechanisms created as a result of this proposed change. No changes are being made to the plant that will introduce any new accident causal mechanisms. The change is solely administrative in nature and does not impact any plant systems that are accident initiators. Therefore, no new accidents or a different accident than previously evaluated is being created.

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

Margin of safety is related to confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. The proposed change is solely administrative in nature and does not affect the performance of the barriers. Consequently, no safety margins will be impacted. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Ms. Lisa F. Vaughn, Duke Power Company LLC, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Branch Chief: Evangelos C. Marinos.

Exelon Generation Company, LLC (EGC), 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. 50–237 and 50–249, Dresden Nuclear Power Station, Units 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. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3, York and Lancaster Counties, Pennsylvania.

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

Date of amendment request: December 15, 2006.

Description of amendment request: The proposed amendment would modify the technical specifications (TSs) by replacing the term ''plant-specific titles'' with ''generic titles'' in TS Section 5.2.1.a, ensuring the TS description is consistent with the EGC Quality Assurance Topical Report (QATR). The proposed amendment will also revise the Peach Bottom TS Section 5.2.1.a, to replace the reference to the Updated Final Safety Analysis Report with reference to the EGC QATR. This will align the Peach Bottom TS wording with the rest of the EGC fleet.

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 is a word replacement in TS 5.2.1, ''Onsite and Offsite Organizations.'' The proposed change involves no changes to plant systems or accident analyses. The proposed change is administrative in nature and, as such, does not affect initiators of analyzed events or assumed mitigation of accidents or transients.

Therefore, the proposed change does not involve any 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.

Creation of the possibility of a new or different kind of accident would require creating one or more new accident precursors. New accident precursors may be created by modifications of plant configuration, including changes in allowable modes of operation. The proposed change does not involve a physical alteration of the plant, add any new equipment, or allow any existing equipment to be operated in a manner different from the present method of operation.

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

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

Response: No.

The proposed change is administrative in nature and has no impact on equipment design or method of operation. There are no changes being made to safety limits or safety system allowable values that would adversely affect plant safety as a result of the proposed change.

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

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

Attorney for licensee: Mr. Bradley J. Fewell, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Michael L. Marshall, Jr.

Exelon Generation Company, LLC (EGC) Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Unit 1, Rock Island County, Illinois

Date of amendment request: January 16, 2007.

Description of amendment request: The proposed amendment revises the values of the safety limit minimum critical power ratio (SLMCPR) in the Quad Cities Nuclear Power Station (QCNPS), Unit 1, Technical Specification (TS) Section 2.1.1, ''Reactor Core SLs [Safety Limits].'' Specifically, the proposed change would require that for QCNPS, Unit 1, the minimum critical power ratio shall be greater than 1.11 for two recirculation loop operation, or greater than 1.13 for single recirculation loop operation. This change is needed to support the next cycle of operation for QCNPS, Unit 1.

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

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

Response: No.

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits have been established consistent with NRC-approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. The proposed change conservatively establishes the SLMCPR for QCNPS, Unit 1, Cycle 20 such that the fuel is protected during normal operation and during plant transients or anticipated operational occurrences (AOOs).

Changing the SLMCPR does not increase the probability of an evaluated accident. The change does not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. Therefore, no individual precursors of an accident are affected.

The proposed change revises the SLMCPR to protect the fuel during normal operation as well as during plant transients or AOOs. Operational limits will be established based on the proposed SLMCPR to ensure that the SLMCPR is not violated. This will ensure that the fuel design safety criterion (i.e., that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and AOOs) is met. Since the proposed change does not affect operability of plant systems designed to mitigate any consequences of accidents, the consequences of an accident previously evaluated are not expected to increase.

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.

Creation of the possibility of a new or different kind of accident requires creating one or more new accident precursors. New accident precursors may be created by modifications of plant configuration, including changes in allowable modes of operation. The proposed change does not involve any plant configuration modifications or changes to allowable modes of operation. The proposed change to the SLMCPR assures that safety criteria are maintained for QCNPS, Unit 1, Cycle 20.

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 SLMCPR provides a margin of safety by ensuring that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and AOOs if the MCPR limit is not violated. The proposed change will ensure the current level of fuel protection is maintained by continuing to ensure that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and AOOs if the MCPR limit is not violated. The proposed SLMCPR values were developed using NRC-approved methods. Additionally, operational limits will be established based on the proposed SLMCPR to ensure that the SLMCPR is not violated. This will ensure that the fuel design safety criterion (i.e., that no more than 0.1% of the rods are expected to be in boiling transition if the MCPR limit is not violated) is met.

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

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

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

Attorney for licensee: Mr. Bradley J. Fewell, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Michael L. Marshall, Jr.

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

Date of amendment request: December 29, 2006.

Description of amendment request: The proposed amendment revises Technical Specification (TS) 3.1.8, ''Scram Discharge Volume (SDV) Vent and Drain Valves,'' to allow a vent or drain line with one inoperable valve to be isolated instead of requiring the valve to be restored to operable status within 7 days.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on February 24, 2003 (68 FR 8637), on possible amendments concerning the consolidated line item implement process (CLIIP), including a model safety evaluation and a model no significant hazards consideration

(NSHC) determination. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on April 15, 2003 (68 FR 18294), as part of the CLIIP. In its application dated December 29, 2006, the licensee affirmed the applicability of the following determination.

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. 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 one valve inoperable instead or requiring the valve to be restored to operable status within 7 days. 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 SDVs 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.

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.

3. The proposed change does not involve a significant reduction in the 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 SDVs is maintained through administrative controls. In addition, the reactor protection system will prevent filling of an 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.

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

Attorney for licensee: David W. Jenkins, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Branch Chief: Michael L. Marshall, Jr.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio, and Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2, Beaver County, Pennsylvania

Date of amendment request: January 11, 2007.

Description of amendment request: The proposed license amendments would modify technical specification (TS) requirements for inoperable snubbers by adding Limiting Condition for Operation (LCO) 3.0.8. The proposed license amendments also modify LCO 3.0.1 to incorporate the addition of LCO 3.0.8. This change is based on the TS Task Force (TSTF) Traveler, TSTF–372, Revision 4. A notice of availability for this TS improvement using the consolidated line item improvement process was published in the **Federal Register** on May 4, 2005.

The Nuclear Regulatory Commission (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 November 24, 2004 (69 FR 68412), and May 4, 2005 (70 FR 23252). The licensee affirmed the applicability of the model NSHC determination in its application dated January 11, 2007.

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

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

The proposed change allows a delay time for entering a supported system technical specification (TS) when the inoperability is due solely to an inoperable snubber if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. Therefore, the probability of an accident previously evaluated is not significantly increased, if at all. The consequences of an accident while relying on allowance provided by proposed LCO 3.0.8 are no different than the consequences of an accident while relying on the TS required actions in effect without the allowance provided by proposed LCO 3.0.8. Therefore the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). Allowing delay times for entering supported system TS when inoperability is due solely to inoperable snubbers, if risk is assessed and managed, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change allows a delay time for entering a supported system TS when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence and the overall TS system safety function would still be available for the vast majority of anticipated challenges. The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in RG 1.177. A bounding risk assessment was performed to justify the proposed TS changes. This application of LCO 3.0.8 is predicated upon the licensee's performance of a risk assessment and the management of plant risk. The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: David W. Jenkins, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Branch Chief: Michael L. Marshall, Jr.

Nine Mile Point Nuclear Station (NMPNS), LLC, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1 (NMP1), Oswego County, New York

Date of amendment request: December 14, 2006.

Description of amendment request: The proposed license amendment would revise the accident source term used in the NMP1 design basis radiological consequence analyses in accordance with 10 CFR 50.67. The revised accident source term replaces the current methodology that is based on TID–14844, ''Calculation of Distance Factors for Power and Test Reactor Sites,'' with the alternative source term (AST) methodology described in Regulatory Guide (RG) 1.183, ''Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.'' The amendment request is for full implementation of the AST as described in RG 1.183, with the exception that TID–14844 will continue to be used as the radiation dose basis for equipment qualification and vital area access.

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.

Adoption of the AST and those plant systems affected by implementing AST do not initiate DBAs [design-basis accidents]. The AST does not affect the design or manner in which the facility is operated; rather, for postulated accidents, the AST is an input to calculations that evaluate the radiological consequences. The AST does not by itself affect the post-accident plant response or the actual pathway of the radiation released from the fuel. It does, however, better represent the physical characteristics of the release, so that appropriate mitigation techniques may be applied. Implementation of the AST has been incorporated in the analyses for the limiting DBAs at NMP1.

The structures, systems and components affected by the proposed change mitigate the consequences of accidents after the accident has been initiated. Application of the AST does result in changes to NMP1 Updated Final Safety Analysis Report (UFSAR) functions (e.g., Liquid Poison system). As a condition of the application of AST, NMPNS is proposing to use the Liquid Poison system to control the suppression pool pH following a LOCA [loss-of-coolant accident]. The proposed changes also revise operability requirements for the secondary containment and certain post-accident filtration systems while handling irradiated fuel that has decayed for greater than 24 hours and during core alterations. These changes have been included within the AST evaluations. These changes do not require any physical changes to the plant. As a result, the proposed changes do not involve a revision to the parameters or conditions that could contribute to the initiation of a DBA discussed in Chapter XV of the NMP1 UFSAR. Since design basis accident initiators are not being altered by adoption of the AST, the probability of an accident previously evaluated is not affected.

Plant-specific AST radiological analyses have been performed and, based on the results of these analyses, it has been demonstrated that the dose consequences of the limiting events considered in the analyses are within the acceptance criteria provided by the NRC for use with the AST. These criteria are presented in 10 CFR 50.67 and Regulatory Guide 1.183. Even though the AST dose limits are not directly comparable to the previously specified whole body and thyroid dose guidelines of General Design Criterion 19 and 10 CFR 100.11, the results of the AST analyses have demonstrated that the 10 CFR 50.67 limits are satisfied. Therefore, it is concluded that adoption of the AST does not involve a significant increase in the consequences of an accident previously evaluated.

Based on the above discussion, it is concluded that 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.

Implementation of AST and the proposed changes do not alter or involve any design basis accident initiators. These changes do not involve any physical changes to the plant and do not affect the design function or mode of operations of systems, structures, or components in the facility prior to a postulated accident. Since systems, structures, and components are operated essentially no differently after the AST implementation, no new failure modes are created by this proposed change.

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 changes proposed are associated with a new licensing basis for analysis of NMP1 DBAs. Approval of the licensing basis change from the original source term to the AST is being requested. The results of the accident analyses performed in support of the proposed changes are subject to revised acceptance criteria. The limiting DBAs have been analyzed using conservative methodologies, in accordance with the guidance contained in Regulatory Guide 1.183, to ensure that analyzed events are bounding and that safety margin has not been reduced. The dose consequences of these limiting events are within the acceptance criteria presented in 10 CFR 50.67 and Regulatory Guide 1.183. Thus, the proposed changes continue to ensure that the doses at the exclusion area boundary and low population zone boundary, as well as in the control room, are within corresponding regulatory criteria.

Therefore, by meeting the applicable regulatory criteria for AST, it is concluded that the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

NRC Acting Branch Chief: John P. Boska.

Nine Mile Point Nuclear Station (NMPNS), LLC, Docket No. 50–410, Nine Mile Point Nuclear Station Unit No. 2 (NMP2), Oswego County, New York

Date of amendment request: January 4, 2007.

Description of amendment request: The proposed license amendment would revise Technical Specification (TS) 3.7.1, ''Service Water (SW) System and Ultimate Heat Sink (UHS),'' as follows: Revise the existing Limiting Condition for Operation (LCO) statement to require four operable SW pumps to be in operation when SW subsystem supply header water temperature is ≤82 °F; add a requirement that five operable SW pumps be in operation when SW subsystem supply header water temperature is >82 °F and ≤84 °F; delete Condition G and the associated Required Actions and Completion Times; revise Surveillance Requirement 3.7.1.3 to increase the maximum allowed SW subsystem supply header water temperature from 82 °F to 84 °F; and modify the requirements for increasing the surveillance frequency as the temperature approaches the limit.

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 eliminates the requirement to perform temperature averaging when the UHS temperature is >82 °F, establishes 84 °F as the design limit for UHS water temperature for operation on a continuous basis, and revises the frequency for verifying that the UHS temperature is within the prescribed limit. The TS currently allow operation with the UHS water temperature temporarily exceeding 82 °F, up to a maximum of 84 °F. The UHS temperature itself is not an initiator of accidents analyzed in the Updated Safety Analysis Report (USAR). Raising the maximum temperature limit and revising the associated surveillance requirement frequency do not involve any plant hardware changes or new operator actions that could serve to initiate an accident. Continuous operation with the elevated UHS temperature may result in a few balance-of-plant equipment high temperature alarms. Operator response to these alarms would be in accordance with established alarm response procedures. In all cases, trip setpoints leading to a reactor scram or a power runback will not be reached, and the likelihood of component failures that could initiate an accident will not be significantly increased.

The potential impact of the proposed change on the ability of the plant to mitigate postulated accidents has been evaluated. These evaluations demonstrate that safety-related systems and components that rely on the UHS as the cooling medium or as a pump suction source are capable of performing their intended safety functions at the higher UHS temperature, and that containment integrity and equipment qualification are maintained. The calculated post-accident dose consequences reflected in the USAR do not directly utilize UHS temperature as an input and thus are not impacted by the proposed change.

Based on the above, the proposed change will have no adverse effect on plant operation or the availability or operation of any accident mitigation equipment. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change will not alter the current plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. The proposed change will not alter the way any structure, system, or component functions and will not cause an adverse effect on plant operation or accident mitigation equipment. The response of the plant and the operators following a design-basis accident is unaffected by the change. The proposed change does not introduce any credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases. Analyses have shown that the design basis heat removal capability of the affected safety-related components is maintained at the increased UHS water temperature limit.

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 margin of safety is determined by the design and qualification of the plant equipment, the operation of the plant within analyzed limits, and the point at which protective or mitigative actions are initiated. The proposed change does not impact these factors. An evaluation of the safety systems has been performed to ensure their safety functions can be met for operation with a UHS water temperature of 84 °F on a continuous basis. Operation with the UHS water temperature temporarily exceeding 82 °F, up to a maximum of 84 °F, is currently allowed. Operating on a continuous basis at the higher UHS temperature represents a slight reduction in design margins in terms of the ability of affected systems to remove accident heat loads. However, the evaluation has demonstrated that the proposed change does not have a significant impact on the capability of the affected systems to perform their safety-related post-accident functions and to mitigate accident consequences. The design limits for the containment and fuel cladding will not be exceeded, and equipment qualification will be maintained. No protection setpoints are affected by the proposed change. The revised frequency for performing the TS surveillance to verify that the UHS temperature is within the prescribed limit will continue to assure that plant operators are aware of and are monitoring increasing UHS temperature trends prior to reaching a value of 82 °F, when a fifth SW pump must be placed in operation. This action is no different than that required by the current TS.

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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1700 K Street, NW., Washington, DC 20006.

NRC Acting Branch Chief: John P. Boska.

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

Date of amendment request: January 29, 2007.

Description of amendment request: The proposed amendment would revise Table 3.3.5.1–1, ''Emergency Core cooling System Instrumentation,'' of the Technical Specifications (TSs) to extend the quarterly surveillance interval from quarterly to a nominal 24-month interval for three low pressure coolant injection loop select logic functions. Consistent with the extended test interval, the licensee also proposed to change the allowable values associated with each of the three logic functions (i.e., response time in seconds). The licensee stated that the quarterly surveillance requirement was inappropriately introduced when the TSs was converted from its previous custom format to the current Improved Technical Specification format by Amendment No. 146. Before the conversion, there was no such quarterly surveillance requirement. Furthermore, the plant was not designed to have these three logic functions tested while on-line.

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the Code of Federal Regulations (10 CFR) Part 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration (NSHC). The NRC staff reviewed the licensee's analysis, and has performed its own analysis as follows:

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

No. The proposed amendment would extend the performance interval from quarterly to a 24-month interval, and change the associated allowable values for the three logic functions. The performance of these surveillances, or the failure to perform, as well as the surveillance finding (*i.e.*, response time in seconds) are not precursors to, and do not affect the probability of, an accident. There is no design or operation change associated with the proposed amendment. Therefore, the proposed amendment does not increase the probability of an accident previously evaluated.

A delay in performing these surveillances would not result in a system being unable to perform its required function. The extended surveillance and associated changed allowable values will not affect the three logic functions to operate as designed. Therefore, the plant systems required to mitigate accidents will remain capable of performing their design function. As a result, the proposed amendment will not lead to any significant change in the consequences of any accident.

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

No. The proposed amendment does not involve a physical alteration of any system, structure, or component (SSC) or a change in the way any SSC is operated. The proposed amendment does not involve operation of any SSCs in a manner or configuration different from those previously recognized or evaluated. No new failure mechanisms will be introduced by the extended surveillance interval and associated allowable values. Thus, 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?

No. The proposed amendment only changes the surveillance interval and associated allowable values for the three logic functions. There will be no modification of any TSs limiting condition for operation, no change to any limit on previously analyzed accidents, no change to how previously analyzed accidents or transients would be mitigated, no change in any methodology used to evaluate consequences of accidents, and no change in any operating procedure or process. The instrumentation and components involved in this proposed amendment have exhibited reliable operation based on the results of their performance during past periodic emergency core cooling system functional testing. 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 the NRC staff's own analysis above, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the proposed amendment 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 Branch Chief: L. Raghavan.

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

Date of amendment request: January 29, 2007.

Description of amendment request: The proposed amendments would revise technical specification (TS) 3.5.3, ''ECCS (Emergency Core Cooling Systems)—Shutdown'' operability requirements for the Safety Injection (SI) subsystem. These revisions will allow the required SI pump to be rendered incapable of injecting into the Reactor Coolant System (RCS) during low temperature (MODE 4) operations due to a single action or automatic signal. The capability of the plant operators to initiate SI flow on a timely basis will be maintained.

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

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

Response: No.

This license amendment request proposes to add a new Note to Technical Specification 3.5.3, ''Emergency Core Cooling System—Shutdown''. This Note will allow the Safety Injection system to be considered operable within the Limiting Condition for Operation requirements while the system is not capable of automatic injection provided it is capable of being manually aligned for injection.

This Emergency Core Cooling System is not an accident initiator, thus the proposed changes do not increase the probability of an accident. The current licensing basis, Technical Specifications and Bases do not require automatic initiation instrumentation for the Emergency Core Cooling System in Mode 4, but rather assume operator action to mitigate an accident. With the proposed Technical Specification and Bases changes, the Emergency Core Cooling System will continue to be operable for manual initiation. Since the changes proposed in this license amendment request do not impact the performance of the system, these changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

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

Response: No.

This license amendment request proposes to add a new Note to Technical Specification 3.5.3, ''Emergency Core Cooling System—Shutdown''. This Note will allow the Safety Injection system to be considered operable within the Limiting Condition for Operation requirements while the system is not capable of automatic injection provided it is capable of being manually aligned for injection.

The changes proposed for the Emergency Core Cooling System Technical Specifications do not change any system operations, maintenance activities or testing requirements. The Limiting Condition for Operation will continue to be met, no new failure modes or mechanisms are created and no new accident precursors are generated by this change. The Technical Specification changes proposed in this license amendment do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

This license amendment request proposes to add a new Note to Technical Specification 3.5.3, ''Emergency Core Cooling System—Shutdown''. This Note will allow the Safety Injection system to be considered operable within the Limiting Condition for Operation requirements while the system is not capable of automatic injection provided it is capable of being manually aligned for injection.

The current licensing basis, Technical Specifications and Bases rely upon operator actions to initiate safety injection to mitigate an accident in Mode 4 and do not require operability of any process instrumentation capable of automatically initiating the Emergency Core Cooling System. With the changes proposed in this license amendment request, the safety injection system will continue to be operable and the plant will continue to rely on operator actions for safety injection initiation. Thus, the Technical Specification changes proposed in this license amendment request do not involve a significant reduction in a margin of safety.

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

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

NRC Acting Branch Chief: P. Milano.

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: October 11, 2006, as supplemented on October 25, November 21, and December 4, 2006.

Description of amendment request: The proposed amendments would increase the SSES 1 and 2 licensed thermal power to 3952 Mega-watts thermal (MWt), which is 20% above the original rated thermal power (RTP) of 3293 MWt, and approximately 13% above the current RTP of 3489 MWt. The proposed amendments would revise the SSES 1 and 2 Operating License and Technical Specifications necessary to implement the increased power level.

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?

Extended Power Uprate

Response: No.

The probability (frequency of occurrence) of Design Basis Accidents occurring is not affected by the increased power level, because Susquehanna continues to comply with the regulatory and design basis criteria established for plant equipment. A probabilistic risk assessment demonstrates that the calculated core damage frequencies do not significantly change due to Constant Pressure Power Uprate (CPPU). Scram setpoints (equipment settings that initiate automatic plant shutdowns) are established such that there is no significant increase in scram frequency due to CPPU. No new challenges to safety-related equipment result from CPPU.

The changes in consequences of postulated accidents, which would occur from 102% of the CPPU (rated thermal power) RTP compared to those previously evaluated, are acceptable. The results of CPPU accident evaluations do not exceed the NRC-approved acceptance limits. The spectrum of postulated accidents and transients has been investigated, and are shown to meet the plant's currently licensed regulatory criteria. In the area of fuel and core design, for example, the Safety Limit Minimum Critical Power Ratio (SLMCPR) and other applicable Specified Acceptable Fuel Design Limits (SAFDLS) are still met. Continued compliance with the SLMCPR and other SAFDLs will be confirmed on a cycle specific basis consistent with the criteria accepted by the NRC.

Challenges to the Reactor Coolant Pressure Boundary were evaluated at CPPU conditions (pressure, temperature, flow, and radiation) were found to meet their acceptance criteria for allowable stresses and overpressure margin.

Challenges to the containment have been evaluated, and the containment and its associated cooling systems continue to meet 10 CFR [Part] 50, Appendix A, Criterion 16, Containment Design; Criterion 38, Containment Heat Removal; and Criterion 50, Containment Design Basis. The increase in the calculated post LOCA [loss-of-coolant accident] suppression pool temperature above the currently assumed peak temperature was evaluated and determined to be acceptable.

Radiological release events (accidents) have been evaluated, and shown to meet the guidelines of 10 CFR 50.67.

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

LPRM [Local Power Range Monitor] Calibration Interval Technical Specification SR [Surveillance Requirement] Frequency Change

Response: No.

The revised surveillance interval continues to ensure that the LPRM signal is adequately calibrated. This change will not alter the basic operation of process variables, structures, systems, or components as described in the SSES FSAR [final safety analysis report], and no new equipment is introduced by the change in LPRM surveillance interval. The performance of the APRM [average power range monitor] and RBM [rod block monitor] systems is not significantly affected by the proposed LPRM surveillance interval increase. Therefore, the probability of accidents previously evaluated is unchanged.

The proposed change results in no change in radiological consequences of the design basis LOCA as currently analyzed for SSES. The consequences of an accident can be affected by the thermal limits existing at the time of the postulated accident, but LPRM chamber exposure has no significant effect on the calculated thermal limits because LPRM accuracy does not significantly deviate with exposure. For the extended calibration interval, the assumption in the safety limit analysis remains valid, maintaining the accuracy of the thermal limit calculation. Therefore, the thermal limit calculation is not significantly affected by LPRM calibration frequency and the consequences of an accident previously evaluated are unchanged.

The change does not affect the initiation of any event, nor does it negatively impact the mitigation of any event. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

RHR [Residual Heat Removal] Service Water System and Ultimate Heat Sink Technical Specification and Methods Change

Response: No.

The proposed changes do not involve any new initiators for any accidents nor do they increase the likelihood of a malfunction of any Structures, Systems or Components (SSCs). Implementation of the subject changes reduces the probability of adverse consequences of accidents previously evaluated, because inclusion of the manual spray array bypass isolation valves and the small spray array isolation valves in the Technical Specifications (TS) increases their reliability to function for safe shutdown. The use of the ANS/ANSI–5.1–1979 decay heat model in the UHS [ultimate heat sink] performance analysis is not relevant to accident initiation, but rather, pertains to the method used to evaluate currently postulated accidents. Its use does not, in any way, alter existing fission product boundaries, and provides a conservative prediction of decay heat. Therefore, the change in decay heat calculational method does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Containment Analysis Methods Change

Response: No.

The use of passive heat sinks, and the ANS/ANSI–5.1–1979 decay heat model are not relevant to accident initiation, but rather, pertain to the method used to evaluate postulated accidents. The use of these elements does not, in any way, alter existing fission product boundaries, and provides a conservative prediction of the containment response to DBA [design-basis accident]-LOCAs. Therefore[,] the Containment Analysis Method Change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Feedwater Pump/Condensate Pump Trip Change

Response: No.

Feedwater pump trips and condensate pump trips rarely occur. A low water level SCRAM on loss of one feedwater pump or one condensate pump is bounded by the loss of all feedwater transient in Final Safety Analysis Report (FSAR) Appendix 15E. A trip of one feedwater pump or a trip of one condensate pump does not result in the loss of all feedwater. The Feedwater Pump / Condensate Pump Trip Change is included in the CPPU Probabilistic Risk Assessment (PRA). The best estimate for the Susquehanna Steam Electric Station (SSES) Core Damage Frequency (CDF) risk increase due to the CPPU is 6E–08 for Unit 1 and 7E–08 for Unit 2 which are in the lower left corner of Region III of Regulatory Guide [sic] (Reference 15) (i.e., very small risk changes). The best estimate for the Large Early Release Frequency (LERF) increase is 1.0E–09/yr for both units which is also in the lower left corner of the Region III range of Regulatory Guide 1.174. Therefore, the Feedwater Pump/Condensate Pump Trip Change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Main Turbine Pressure Regulation System

Response: No.

Technical Specification 3.7.8 does not directly or indirectly affect any plant system, equipment, component, or change the process used to operate the plant. Technical Specification 3.7.8 would ensure acceptable performance, since it would establish requirements for adhering to the appropriate thermal limits, depending on the operability of the main turbine pressure regulation system. Use of the appropriate limits assures that the appropriate safety limits will not be exceeded during normal or anticipated operational occurrences. Thus, Technical Specification 3.7.8 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?

Extended Power Uprate

Response: No.

Equipment that could be affected by EPU has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified. The full spectrum of accident considerations has been evaluated and no new or different kind of accident has been identified. CPPU uses developed technology and applies it within capabilities of existing or modified plant safety related equipment in accordance with the regulatory criteria (including NRC approved codes, standards and methods). No new accidents or event precursors have been identified.

The SSES TS require revision to implement EPU. The revisions have been assessed and it was determined that the proposed change will not introduce a different accident than that previously evaluated. Therefore[,] the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

LPRM Calibration Interval Technical Specification SR Frequency Change

Response: No.

The proposed change will not physically alter the plant or its mode of operation. The performance of the APRM and RBM systems is not significantly affected by the proposed LPRM surveillance interval increase. As such, no new or different types of equipment will be installed and the basic operation of installed equipment is unchanged. The methods of governing plant operation and testing are consistent with current safety analysis assumptions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

RHR Service Water System and Ultimate Heat Sink Technical Specification and Methods Change

Response: No.

The subject changes apply Technical Specification controls to new UHS manual bypass isolation valves and the existing small spray array isolation valves. The design functions of the systems are not affected.

The addition of manually operated valves in the system, operational changes and the Technical Specification changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The use of the ANS/ANSI–5.1–1979 decay heat model is not relevant to accident initiation, but rather pertains to the method used to evaluate currently postulated accidents. The use of this analytical tool does not involve any physical changes to plant structures or systems, and does not create a new initiating event for the spectrum of events currently postulated in the FSAR. Further, it does not result in the need to postulate any new accident scenarios. Therefore[,] the decay heat calculational method change does not create the possibility of a new or different kind of accident from any accident previously evaluated[.]

Containment Analysis Methods Change

Response: No.

The use of passive heat sinks and the ANS/ANSI–5.1–1979 decay heat model are not relevant to accident initiation, but pertain to the method used to evaluate currently postulated accidents. The use of these analytical tools does not involve any physical changes to plant structures or systems, and does not create a new initiating event for the spectrum of events currently postulated in the FSAR. Further, they do not result in the need to postulate any new accident scenarios. Therefore, the Containment Analysis Method Change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Feedwater Pump/Condensate Pump Trip Change

Response: No.

The occurrence of a reactor SCRAM is already considered in the current licensing basis and is not an accident. A SCRAM resulting from the trip of a feedwater pump or a condensate pump is bounded by a loss of all feedwater event. The loss of all feedwater transient is already considered in the plant licensing basis. The SCRAM due to the feedwater or condensate pump trip does not change the results of the loss of all feedwater transient in any way. Therefore, the Feedwater Pump/Condensate Pump Trip Change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Main Turbine Pressure Regulation System

Response: No.

Technical Specification 3.7.8 will not directly or indirectly affect any plant system, equipment, or component and therefore does not affect the failure modes of any of these items. Thus, Technical Specification 3.7.8 does not create the possibility of a previously unevaluated operator error or a new single failure.

Therefore, Technical Specification 3.7.8 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?

Extended Power Uprate

Response: No.

The CPPU affects only design and operational margins. Challenges to the fuel, reactor coolant pressure boundary, and containment were evaluated for CPPU conditions. Fuel integrity is maintained by meeting existing design and regulatory limits. The calculated loads on affected structures, systems and components, including the reactor coolant pressure boundary, will remain within their design allowables for design basis event categories. No NRC acceptance criterion is exceeded. Because the SSES configuration and responses to transients and postulated accidents do not result in exceeding the presently approved NRC acceptance limits, the proposed changes do not involve a significant reduction in a margin of safety.

LPRM Calibration Interval Technical Specification Change

Response: No.

The proposed change has no impact on equipment design or fundamental operation and there are no changes being made to safety limits or safety system allowable values that would adversely affect plant safety as a result of the proposed change. The performance of the APRM and RBM systems is not significantly affected by the proposed LPRM surveillance interval increase. The margin of safety can be affected by the thermal limits existing prior to an accident; however, uncertainties associated with LPRM chamber exposure have no significant effect on the calculated thermal limits. For the extended calibration interval, the assumption in the safety limit analysis remains valid, maintaining the accuracy of the thermal limit calculation.

Since the proposed change does not affect safety analysis assumptions or initial conditions, the margin of safety in the safety analyses are maintained. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

RHR Service Water System and Ultimate Heat Sink Technical Specification and Methods Change

Response: No.

Implementation of the subject changes does not significantly reduce the margin of safety since these changes add components and Technical Specification controls for the components not currently addressed in the Technical Specifications. These changes increase the reliability of the affected components/systems to function for safe shutdown.

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

The ANS/ANSI–5.1–1979 model provides a conservative prediction of decay heat. The use of this element is consistent with current industry standards, and has been previously accepted by the staff for use in containment analysis by other licensees, as described in GE Nuclear Energy. ''Constant Pressure Power Uprate,'' Licensing Topical Report NEDC–33004P–A, Revision 4, dated July 2003; and the letter to Gary L. Sozzi (GE) from Ashok Thandani (NRC) on the Use of the SHEX Computer Program and ANSI/ANS 5.1–1979, ''Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis,'' July 13, 1993. Therefore, the decay heat calculational method change does not involve a significant reduction in the margin of safety.

Containment Analysis Methods Change

Response: No.

The use of passive heat sinks and the ANS/ANSI–5.1–1979 decay heat model are realistic phenomena, and provide a conservative prediction of the plant response to DBA–LOCAs. The use of these elements is consistent with current industry standards, and has been previously accepted by the staff for other licensees, as described in GE Nuclear Energy: ''Constant Pressure Power Uprate,'' Licensing Topical Report NEDC–33004P–A, Revision 4, dated July 2003; the letter to Gary L. Sozzi (GE) from Ashok Thandani (NRC) on the Use of the SHEX Computer Program; and ANSI/ANS 5.1–1979, ''Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis,'' July 13, 1993. Therefore the Containment Analysis Method Change does not involve a significant reduction in [a] margin of safety.

Feedwater Pump/Condensate Pump Trip Change

Response: No.

A low water level SCRAM on loss of one feedwater pump or one condensate pump is bounded by the loss of all feedwater transient in FSAR Appendix 15E. The loss of all feedwater transient is a non-limiting event that does not contribute to the setting of the fuel safety limits. Consequently, a SCRAM resulting from a feedwater pump or condensate pump trip does not reduce the margin to fuel safety limits. Therefore, the potential for a SCRAM resulting from a feedwater pump trip or a condensate pump trip does not involve a significant reduction in [a] margin of safety.

Main Turbine Pressure Regulation System

Since Technical Specification 3.7.8 does not alter any plant system, equipment, component, or processes used to operate the plant, the proposed change will not jeopardize or degrade the function or operation of any plant system or component governed by Technical Specifications. Technical Specification 3.7.8 preserves the margin of safety by establishing requirements for adhering to the appropriate thermal limits.

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 (Acting) Branch Chief: Douglas V. Pickett.

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

Date of amendment request: February 2, 2007.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) LCO 3.10.1 to expand its scope to include provisions for temperature excursions greater than 212 degrees F as a consequence of scram time testing initiated in conjunction with an inservice leak or hydrostatic test. During these tests and with temperature greater than 212 degrees F, operational conditions are considered to be in Mode 4.

The NRC staff issued a notice of availability of a model safety evaluation and model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on October 27, 2006 (71 FR 63050). The licensee affirmed the applicability of the model NSHC determination in its application dated February 2, 2007.

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

Technical Specifications currently allow for operation at greater than 212 deg F while imposing MODE 4 requirements in addition to the secondary containment requirements required to be met. Extending the activities that can apply this allowance will not adversely impact the probability or consequences of an accident previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Technical Specifications currently allow for operation at greater than 212 deg F while imposing MODE 4 requirements in addition to the secondary containment requirements required to be met. No new operational conditions beyond those currently allowed by LCO 3.10.1 are introduced. The changes do 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. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are 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 accident previously evaluated.

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

Technical Specifications currently allow for operation at greater than 212 deg F while imposing MODE 4 requirements in addition to the secondary containment requirements required to be met. Extending the activities that can apply this allowance will not adversely impact any margin of safety. Allowing completion of inspections and testing and supporting completion of scram time testing initiated in conjunction with an inservice leak or hydrostatic test prior to power operation results in enhanced safe operations by eliminating unnecessary maneuvers to control reactor temperature and pressure. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: January 12, 2007.

Description of amendment request: The proposed amendment would revise the steam generator (SG) program requirements in the Sequoyah (SQN) Unit 2 Technical Specifications (TSs) to allow use of an SG voltage-based repair criteria probability of detection (POD) method using plant-specific SG tube inspection results. The proposed POD method is referred to as the probability of prior cycle detection (POPCD) method.

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

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

Response: No. The use of a revised SG voltage-based repair criteria POD method, the POPCD method, to determine the BOC [beginning of cycle] indication voltage distribution for the SQN Unit 2 operational assessments does not increase the probability of an accident. Based on industry and plant-specific bobbin detection data for ODSCC [outside diameter stress corrosion cracking] within the SG tube support plate (TSP) region, large voltage bobbin indications which individually can challenge structural or leakage integrity can be detected with near 100 percent certainty. Since large voltage outside diameter stress corrosion cracking ODSCC bobbin indications within the SG TSP can be detected, they will not be left in service, and therefore these indications should not be included in the voltage distribution for the purpose of operational assessments. The POPCD method improves the estimate of potentially undetected indications for operational assessments, but does not directly affect the inspection results. Since large voltage indications are detected, they will not result in an increase in the probability of SG tube rupture accident or an increase in the consequences of a tube rupture or main steam line break (MSLB) accident.

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

Response: No.

The use of the POPCD method is associated with numerical predictions of probabilities for the steam generator tube rupture (SGTR) accident. Since the SGTR accident is considered in SQN's Updated Final Safety Analysis Report, there is no possibility to create a design basis accident that has not been previously evaluated. Therefore, the proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

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

Response: No. The use of the POPCD method to determine the BOC voltage distribution for the SQN Unit 2 operational assessments does not involve a significant reduction in a margin of safety. The applicable margin of safety potentially impacted is the SG tube structural and leakage criteria. Based on industry and plant-specific bobbin detection data for ODSCC within the SG TSP region, large voltage bobbin indications that can individually challenge structural or leakage integrity can be detected with near 100 percent certainty and will not be left in service. Therefore, these indications should not be included in the voltage distribution for the purposes of operational assessments. Since these large voltage indications are detected, they will not result in a significant increase in the actual EOC [end of cycle] leakage for a MSLB accident or the actual EOC probability of burst. The POPCD method approach to POD considers the potential for missing indications that might challenge structural or leakage integrity by applying the POPCD data from successive inspections. If a large indication was missed in one inspection, it would continue to grow until detected in a later inspection. Accordingly, there is no significant increase in the margin of safety.

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

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

NRC Branch Chief: Brenda Mozafari (Acting).

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

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

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

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

Date of amendment request: February 2, 2007.

Description of amendment request: The proposed amendment would revise Technical Specification 3.6.1.7, ''Suppression Chamber-to-Drywell Vacuum Breakers,'' to allow a one-time extension to the current closure verification surveillance requirement for one of two redundant disks in one of nine vacuum breakers until reliable position indication can be restored in the main control room during the next refueling outage (R–18), which is scheduled to begin on May 12, 2007.

Date of publication of individual notice in ***Federal Register:*** February 12, 2007 (72 FR 6606).

Expiration date of individual notice: February 26, 2007.

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

Date of application for amendment: December 27, 2006.

Brief description of amendment: The proposed amendment would revise Limiting Condition for Operation 3.14.A to adopt the Technical Specification Task Force 484, Revision 0, ''Use of Technical Specification 3.10.1 for Scram Time Testing Activities.''

Date of publication of individual notice in ***Federal Register:*** February 20, 2007 (72 FR 7776).

Expiration date of individual notice: March 22, 2007 (public comments) and April 23, 2007 (hearing requests).

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

Date of application for amendment: January 15, 2007.

Brief description of amendment: The amendment request supercedes the previously submitted license amendment request dated April 12, 2006, proposing new Pressure-Temperature (PT) curves and to extend the applicability of current PT limits expressed in Technical Specification Figures 3.6.1, 3.6.2, and 3.6.3 through the end of operating cycle 18.

Date of publication of individual notice in ***Federal Register***: February 12, 2007 (72 FR 6609).

Expiration date of individual notice: March 14, 2007 (public comments) and April 13, 2007 (hearing requests).

PSEG Nuclear LLC, Docket No. 50–272, Salem Nuclear Generating Station, Unit No. 1, Salem County, New Jersey

Date of amendment request: January 18, 2007.

Brief description of amendment request: The amendment request proposes a one-time change to the Technical Specifications (TSs) regarding the steam generator (SG) tube inspection and repair required for the portion of the SG tubes passing through the tubesheet region. Specifically, for Salem Unit No. 1 refueling outage 18 (planned for spring 2007) and the subsequent operating cycle, the proposed TS changes would limit the required inspection (and repair if degradation is found) to the portions of the SG tubes passing through the upper 17 inches of the approximate 21-inch tubesheet region.

Date of publication of individual notice in ***Federal Register:*** January 25, 2007 (72 FR 3427).

Expiration date of individual notice: February 26, 2007 (public comments) and March 26, 2007 (hearing requests).

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 51.22(b) and has made a determination based on that assessment, it is so indicated.

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

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

Date of application for amendments: September 28, 2006.

Brief description of amendments: The amendments revised Technical Specification (TS) requirements for mode change limitations in Limiting Condition for Operation (LCO) 3.0.4 and Surveillance Requirement 3.0.4 to adopt the provisions of Industry/TS Task Force (TSTF) Traveler number TSTF–359, ''Increase Flexibility in Mode Restraints.'' The amendments also revised TS Example 1.4–1 to reflect the changes made to LCO 3.0.4 and to be consistent with TSTF–485, which has been incorporated into the Standard Technical Specifications Revision 3.1.

Date of issuance: February 21, 2007.

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

Amendment Nos.: Unit 1–165, Unit 2—165, Unit 3—165.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Operating License and Technical Specifications.

Date of initial notice in ***Federal Register***: November 7, 2006 (71 FR 65140). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 21, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 14, 2006.

Brief description of amendments: The amendments revised Technical Specification (TS) requirements in the Limiting Condition for Operation for TS 3.6.3, ''Containment Isolation Valves,'' and associated Actions and Surveillance Requirements to allow for a blind flange to be used for containment isolation in each of the two flow paths of the 42-inch refueling purge valves in Modes 1 through 4, without remaining in TS 3.6.3 Condition D.

Date of issuance: February 22, 2007.

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

Amendment Nos.: Unit 1–166, Unit 2–166, Unit 3–166.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Operating License and Technical Specifications.

Date of initial notice in ***Federal Register***: March 14, 2006 (71 FR 13171).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 22, 2007.

No significant hazards consideration comments received: No.

Carolina Power & Light Company, Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of application for amendments: April 26, 2006.

Brief Description of amendments: Revised the Technical Specification (TS) requirements for inoperable snubbers by adding Limiting Condition for Operation 3.0.8.

Date of issuance: February 15, 2007.

Effective date: February 15, 2007, implement within 90 days.

Amendment Nos.: 241 and 269.

Renewed Facility Operating License Nos. DPR–71 and DPR–62: Amendments change the TSs.

Date of initial notice in ***Federal Register***: June 6, 2006 (71 FR 32603).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 15, 2007.

Duke Power Company LLC, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: April 11, 2006, as supplemented November 29, 2006.

Brief description of amendments: The amendments revised the Technical Specifications (TSs) related to steam generator tube integrity. The changes are consistent with the consolidated line-item improvement process, Nuclear Regulatory Commission's approved Technical Specification Task Force (TSTF) Standard Specification Change Traveler, TSTF–449, Revision 4, ''Steam Generator Tube Integrity.''

Date of issuance: March 1, 2007.

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

Amendment Nos.: 237, 218.

Renewed Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the licenses and the technical specifications.

Date of initial notice in ***Federal Register***: December 5, 2006 (71 FR 70557) The supplement dated November 29, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 1, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 20, 2006.

Brief description of amendment: The amendment removed ANO–2 reactor coolant structural integrity requirements contained in TS 3.4.10.1. The TS change is consistent with NUREG–1432, ''Standard Technical Specifications Combustion Engineering Plants,'' Revision 3.1. The Bases for TS 3.4.10.1 will be deleted and performed under the ANO–2 TS Bases Control Program, and is not included with the submittal. The amendment also renumbers TS pages 3/4 4–22a, 23, 23a, and 23b as TS pages 3/4 4–23, 24, 25, and 26, respectively.

Date of issuance: March 1, 2007.

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

Amendment No.: 270.

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

Date of initial notice in ***Federal Register***: May 9, 2006 (71 FR 26999). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 2007.

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: November 1, 2006.

Brief description of amendment: The amendment modified technical specification requirements for inoperable snubbers by adding Limiting Condition of Operation 3.0.8 using the Consolidated Line Item Improvement Process.

Date of issuance: February 20, 2007.

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

Amendment No: 171.

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

Date of initial notice in ***Federal Register***: December 5, 2006 (71 FR 70558). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 20, 2007.

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: November 13, 2006.

Brief description of amendment: The amendment revised Grand Gulf Nuclear Station, Unit 1, Technical Specification (TS) Limiting Condition of Operation 3.10.1, and the associated TS Bases, to expand its scope to include provisions for temperature excursions greater than 200 °F as a consequence of inservice leak and hydrostatic testing, and as a consequence of scram time testing initiated in conjunction with an inservice leak or hydrostatic test, while considering operational conditions to be in MODE 4.

Date of issuance: February 21, 2007.

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

Amendment No: 172.

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

Date of initial notice in ***Federal Register***: December 19, 2006 (71 FR 75993). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 21, 2007.

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: May 8, 2006, as supplemented by letter dated November 16, 2006.

Brief description of amendment: The change added an NRC-approved topical report to the analytical methods referenced in Technical Specification Section 5.6.5, ''Core Operating Limits Report (COLR).''

Date of issuance: February 22, 2007.

Effective date: As of the date of issuance and shall be implemented prior to Cycle 16 operation.

Amendment No: 173.

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

Date of initial notice in ***Federal Register****:* June 20, 2006 (71 FR 35458). The supplement dated November 16, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 22, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 31, 2006, as supplemented by letter dated August 30, 2006.

Brief description of amendment: The amendments revise the Technical Specifications (TSs) associated with steam generator tube integrity consistent with Revision 4 to the TS Task Force (TSTF) Standard Technical Specification Change Document TSTF–449, ''Steam Generator Tube Integrity.''

Date of issuance: February 20, 2007.

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

Amendment No.: 251 and 233.

Facility Operating License Nos. DPR–26 and DPR–64: The amendment revised the License and the TSs.

Date of initial notice in ***Federal Register****:* August 1, 2006 (71 FR 43531). The August 30, 2006, supplement provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 20, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 4, 2006.

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

Date of issuance: February 15, 2007.

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

Amendment Nos.: 181/168.

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Technical Specifications and License.

Date of initial notice in ***Federal Register****:* August 15, 2006 (71 FR 46933). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 15, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 25, 2005, as supplemented by letters dated November 11, 2005, April 19, July 10, 2006, September 1, October 24, December 7, 2006, and February 1, 2007.

Brief description of amendments: The amendment converts the current Technical Specifications to the Improved Technical Specifications (ITSs) format and relocates certain requirements to other licensee-controlled documents. The ITSs are based on NUREG–1431, ''Standard Technical Specifications—Westinghouse Plants,'' Revision 2, with the Technical Specification Task Force changes to make the Beaver Valley Power Station Unit Nos. 1 and 2 (BVPS–1 and 2) ITS more consistent with Revision 3; the Commission's Final Policy Statement, ''NRC Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors,'' dated July 22, 1993 (58 FR 39132); and 10 CFR 50.36, ''Technical specifications.'' The purpose of the conversion is to provide clearer and more readily understandable requirements in the TSs for BVPS–1 and 2 to ensure safe operation. In addition, the amendment includes a number of issues that were considered beyond the scope of NUREG–1431.

Date of issuance: February 1, 2007.

Effective date: As of the date of issuance, and shall be implemented within 150 days.

Amendment Nos.: 278 and 161.

Facility Operating License Nos. DPR–66 and NPF–73: The amendment revised the License and the Technical Specifications.

*Date of initial notice in **Federal Register**:* March 22, 2006 (71 FR 14554). The letters dated November 11, 2005, April 19, July 10, 2006, September 1, October 24, December 7, 2006, and February 1, 2007, 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 February 1, 2007.

No significant hazards consideration comments received: No.

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

Date of amendment request: April 28, 2006.

Description of amendment request: The amendment revised the Seabrook Technical Specifications (TSs) Limiting Condition for Operation 3.0.4 and Surveillance Requirement (SR) 4.0.4 to adopt the provisions of Industry/TS Task Force (TSTF) change TSTF–359, Revision 9, ''Increased Flexibility in Mode Restraints.'' TSTF–359 is part of the consolidated line item improvement process. Specifically, the proposed change allows, for systems and components, mode changes into a TS condition that has a specific required action and completion time.

Date of issuance: February 9, 2007.

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

Amendment No.: 114.

Facility Operating License No. NPF–86: The amendment revised the TSs.

*Date of initial notice in **Federal Register**:* July 5, 2006 (71 FR 38182). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 9, 2007.

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: October 23, 2006.

Brief description of amendments: The amendments to the Technical Specifications (TSs) eliminate the use of the defined term CORE ALTERATIONS in the TSs.

Date of issuance: February 15, 2007.

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

Amendment Nos.: 224 & 230.

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

*Date of initial notice in **Federal Register**:* December 5, 2006 (71 FR 70562). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 15, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 13, 2006.

Brief description of amendments: The amendments revise Prairie Island Nuclear Generating Plant, Units 1 and 2, Technical Specifications (TS) to change the wording in TS 3.0, ''Surveillance Requirement (SR) Applicability'' and change format and titles in TS 5.0, ''Administrative Controls.'' The proposed changes improve the TS usability, conformance with the industry standard, NUREG–1431, ''Standard Technical Specifications, Westinghouse Plants,'' Revision 3.0 and accuracy.

Date of issuance: February 13, 2007.

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

Amendment Nos.: 176 and 166.

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

*Date of initial notice in **Federal Register**:* April 11, 2006 (71 FR 18375). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 13, 2007.

No significant hazards consideration comments received: No.

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

Date of amendment request: November 13, 2006.

Brief description of amendment: The amendment relocated the requirements of Technical Specification (TS) 2.22, ''Toxic Gas Monitors,'' and TS Table 3–3, Item 29, to the Fort Calhoun Station, Unit No. 1, Updated Safety Analysis Report.

Date of issuance: February 28, 2007.

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

Amendment No.: 248.

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

*Date of initial notice in **Federal Register**:* December 19, 2006 (71 FR 75996). The Commission's related evaluation of the amendment is contained in a safety evaluation dated February 28, 2007.

No significant hazards consideration comments received: No.

PPL Susquehanna, LLC, Docket No. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2 (SSES 1 and 2), Luzerne County, Pennsylvania

Date of application for amendments: April 28, 2006.

Brief description of amendments: The amendments revise the SSES 1 and 2 Technical Specifications 3.1.7, ''Standby Liquid Control (SLC) System,'' to modify the SLC system for single loop pump operation and the use of enriched sodium pentaborate solution.

Date of issuance: February 28, 2007.

Effective date: As of the date of issuance and to be implemented prior to the startup following the SSES 1 Spring 2008 15th refueling outage and SSES 2 Spring 2007 13th refueling outage for Units 1 and 2, respectively.

Amendment Nos.: 240 and 217.

Facility Operating License Nos. NPF–14 and NPF–22: The amendments revised the TSs and license.

Date of initial notice in ***Federal Register****:* August 15, 2006 (71 FR 46936). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 28, 2007.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket Nos. 50–259 Browns Ferry Nuclear Plant, Unit 1, Limestone County, Alabama

Date of application for amendment: May 1, 2006 (TS–455), as supplemented by letters dated September 1, and November 6, 2006.

Brief description of amendment: The amendment revises the numeric values of the safety limit critical power ratio (SLMCPR) in the Technical Specification (TS) Section 2.1.1.2 for one and two reactor recirculation loop operation to incorporate the results of the Cycle 7 SLMCPR analysis.

Date of issuance: February 6, 2007.

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

Amendment No.: 267.

Facility Operating License Nos. DPR–33: Amendment revised the TSs.

Date of initial notice in ***Federal Register****:* August 15, 2006 (71 FR 46937). The supplements dated September 1, and November 6, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 6, 2007.

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: August 22, 2005, as supplemented by letters dated September 18 and October 23, 2006.

Brief description of amendments: The amendments revised the Final Safety Evaluation Report Sections 1, 6, and 15. The changes reflect the licensee's adoption of Nuclear Regulatory Commission's Regulatory Guide 1.195, ''Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Reactors,'' for calculating radiological consequences and replacement of steam generators for Comanche Peak Steam Electric Station, Unit 1, in the spring of 2007.

Date of issuance: February 20, 2007.

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

Amendment Nos.: 130/130.

Facility Operating License Nos. NPF–87 and NPF–89: The amendments revised the Final Safety Analysis Report and Facility Operating Licenses.

Date of initial notice in ***Federal Register****:* November 8, 2005 (70 FR 67754). The supplements dated September 18 and October 23, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 20, 2007.

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: February 21, 2006, as supplemented by letters dated September 12 and December 14, 2006.

Brief description of amendments: The amendments increased the allowable values (AVs) for steam generator (SG) water level trip setpoints and the required minimum SG secondary side water inventory in shutdown modes for the replacement SGs in Comanche Peak Steam Electric Station (CPSES), Unit 1. For CPSES Unit 2, the corresponding AVs and the SG secondary water inventory in the current TSs remain unchanged since the existing SGs in Unit 2 will continue to be used.

Date of issuance: February 20, 2007.

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

Amendment Nos.: NPF–87—131; NPF–89—131.

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

Date of initial notice in ***Federal Register****:* June 6, 2006 (71 FR 32609). The supplements dated September 12 and December 14, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 20, 2007.

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: December 16, 2005, as supplemented by letters dated August 31 and September 29, 2006.

Brief description of amendments: The amendments revised Technical Specifications (TSs) 1.1 and 5.6.6 consistent with the Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–419, ''Revise PTLR [Pressure Temperature Limits Report] Definition and References in ISTS [Improved Standard Technical Specification] 5.6.6.

Date of issuance: February 22, 2007.

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

Amendment Nos.: NPF–87–132 and NPF–89–132.

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

Date of initial notice in ***Federal Register****:* March 14, 2006 (71 FR

13182). The supplements dated August 31 and September 29, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 22, 2007.

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: December 12, 2005.

Brief description of amendments: The amendments revise the Technical Specification (TS) Surveillance Requirements (SRs) 3.3.1.2 and 3.3.1.3, ''Reactor Trip System (RTS) Instrumentation.'' The license amendment request is based on Technical Specification Task Force (TSTF) Traveler, TSTF–371–A, Revision 1, ''NIS [Nuclear Instrumentation System] Power Range Channel Daily SR TS Change to Address Low Power Decalibration.'' TSTF–371–A, Revision 1, revised the requirements for performing a daily surveillance adjustment of the power range channel(s) to address industry concern that compliance with SR 3.3.1.2 and SR 3.3.1.3 may result in a non-conservative channel calibration during reduced-power operations. The changes resolved the issue of non-conservatism.

Date of issuance: February 26, 2007.

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

Amendment Nos.: NPF–87–133, NPF–89–133.

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

*Date of initial notice in **Federal Register**:* March 28, 2006 (71 FR 15490).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 26, 2007.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 30, 2006, as supplemented by letters dated November 22 and December 19, 2006.

Brief description of amendment: The amendment revised Surveillance Requirements (SRs) 3.5.2.8 and 3.6.7.1 due to (1) the future replacement of the existing containment recirculation sump suction inlet trash racks and screens with strainers, (2) the resulting relocation of the recirculation fluid pH control (RFPC) system from the sump, and (3) the removal of details from SR 3.6.7.1, including the relocation of the name of the RFPC chemical to a license condition in Appendix C to the license. The modifications will be done in the refueling outage scheduled for the spring of 2007. The amendment also deleted the footnote to the frequency for SR 3.5.2.5 because it is no longer applicable.

Date of issuance: February 21, 2007.

Effective date: As of its date of issuance, and shall be implemented prior to entry into Mode 4 during the plant startup from the refueling outage scheduled for the spring of 2007.

Amendment No.: 180.

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

*Date of initial notice in **Federal Register***: August 15, 2006 (71 FR 46940). The supplemental letters dated November 22 and December 19, 2006, 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 published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 21, 2007.

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

[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.

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).

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

Date of amendment request: February 2, 2007.

Description of amendment request: The amendment revised Technical Specification 3.6.1.7, ''Suppression Chamber-to-Drywell Vacuum Breakers,'' to allow a one-time extension to the current closure verification surveillance requirement for one of two redundant disks in one of nine vacuum breakers until reliable position indication can be restored in the main control room during the next refueling outage (R–18), which is scheduled to begin on May 12, 2007.

Date of issuance: February 27, 2007.

Effective date: As of its date of issuance and shall be implemented within 14 days from the date of issuance.

Amendment No.: 202.

Facility Operating License No.: NPF–21: Amendment revises the technical specifications and license.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. 72 FR 6606, published February 12, 2007. The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided an opportunity to request a hearing within 60 days after the date of publication of the notice, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated February 27, 2007.

Attorney for licensee: William A. Horin, Esq., Winston & Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

NRC Branch Chief: David Terao.

Dated at Rockville, Maryland, this 2nd day of March 2007.

For the Nuclear Regulatory Commission.

Michael C. Cheok,

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

[FR Doc. E7–4251 Filed 3–12–07; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Proposed Collection; Comment Request

Upon Written Request, Copies Available From: Securities and Exchange Commission, Office of Filings and Information Services, Washington, DC 20549.

Extension:
Rule 17j–1, SEC File No. 270–239, OMB Control No. 3235–0224.

Notice is hereby given that, pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 350l–3520), the Securities and Exchange Commission (the ''Commission'') is soliciting comments on the collection of information summarized below. The Commission plans to submit this existing collection of information to the Office of Management and Budget (''OMB'') for extension and approval.

Conflicts of interest between investment company personnel (such as portfolio managers) and their funds can arise when these persons buy and sell securities for their own accounts (''personal investment activities''). These conflicts arise because fund personnel have the opportunity to profit from information about fund transactions, often to the detriment of fund investors. Beginning in the early 1960s, Congress and the Securities and Exchange Commission (''Commission'') sought to devise a regulatory scheme to effectively address these potential conflicts. These efforts culminated in the addition of section 17(j) to the Investment Company Act of 1940 (the ''Investment Company Act'') (15 U.S.C. 80a–17(j)) in 1970 and the adoption by the Commission of rule 17j–1 (17 CFR 270.17j–1) in 1980.[1] The Commission proposed amendments to rule 17j–1 in 1995 in response to recommendations made in the first detailed study of fund policies concerning personal investment activities by the Commission's Division of Investment Management since rule 17j–1 was adopted. Amendments to rule 17j–1, which were adopted in 1999, enhanced fund oversight of personal investment activities and the board's role in carrying out that oversight.[2] Additional amendments to rule 17j–1 were made in 2004, conforming rule 17j–1 to rule 204A–1 under the Investment Advisers Act of 1940 (15 U.S.C. 80b), avoiding duplicative reporting, and modifying certain definitions and time restrictions.[3]

Section 17(j) makes it unlawful for persons affiliated with a registered investment company(''fund'') or with the fund's investment adviser or principal underwriter (each a ''17j–1 organization''), in connection with the purchase or sale of securities held or to be acquired by the investment company, to engage in any fraudulent, deceptive, or manipulative act or practice in contravention of the Commission's rules and regulations. Section 17(j) also authorizes the Commission to promulgate rules requiring 17j–1 organizations to adopt codes of ethics.

In order to implement section 17(j), rule 17j–1 imposes certain requirements on 17j–1 organizations and ''Access Persons'' [4] of those organizations. The

[1] Prevention of Certain Unlawful Activities with Respect to Registered Investment Companies, Investment Company Act Release No. 11421 (Oct. 31, 1980) (45 FR 73915 (Nov. 7, 1980)).

[2] Personal Investment Activities of Investment Company Personnel, Investment Company Act Release No. 23958 (Aug. 20, 1999) (64 FR 46821–01 (Aug. 27, 1999)).

[3] Investment Adviser Codes of Ethics, Investment Advisers Act Release No. 2256 (Jul. 2, 2004) (66 FR 41696 (Jul. 9, 2004)).

[4] Rule 17j–1(a)(1) defines an ''access person'' as ''Any advisory person of a Fund or of a Fund's investment adviser. If an investment adviser's primary business is advising Funds or other advisory clients, all of the investment adviser's directors, officers, and general partners are presumed to be Access Persons of any Fund advised by the investment adviser. All of a Fund's directors,

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