In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to *dkw@nrc.gov*.

Dated: April 6, 2006.

R. Michelle Schroll,

Office of the Secretary.

[FR Doc. 06–3486 Filed 4–7–06; 12:12 pm]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from March 17, 2006 to March 30, 2006. The last biweekly notice was published on March 28, 2006 (71 FR 15479).

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Date of amendment request: February 7, 2006.

Description of amendment request: The proposed amendments would increase the allowed outage time from 72 hours to 7 days for the inoperability of the steam supply to the turbine-driven auxiliary feedwater (AFW) pump or the inoperability of the turbine-driven AFW pump under certain operating mode restrictions.

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

Criterion 1: Does the Proposed Amendment Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated?

Response: No.

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

The proposed amendment to MPS 2 and 3 [Millstone Power Station, Unit Nos. 2 and 3] TS [Technical Specification] 3.7.1.2 permits a 7 day allowed outage time for the inoperability of the necessary steam supply to the turbine-driven AFW pump in Modes 1, 2, and 3, or for the inoperability of the turbine-driven AFW pump if the inoperability occurs in Mode 3 following a refueling outage, if Mode 2 had not been entered. Extending the allowed outage time does not involve a significant increase in the probability or consequences of an accident previously evaluated because: 1) The proposed amendment does not represent a change to the system design, 2) the proposed amendment does not prevent the safety function of the AFW [system] from being performed since the redundant trains are required to be operable, 3) the proposed amendment does not alter, degrade, or prevent action described or assumed in any accident described in the MPS 2 and 3 FSARs [final safety analysis reports] from being performed since the other trains of AFW are required to be operable, 4) the proposed amendment does not alter any assumptions previously made in evaluating radiological consequences, and 5) the proposed amendment does not affect the integrity of any fission product barrier. No other safety related equipment is affected by the proposed change. Therefore, this proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2: Does the Proposed Amendment Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated?

Response: No.

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

The proposed amendment to MPS 2 and 3 TS 3.7.1.2 would allow a 7 day allowed outage time for the inoperability of the necessary steam supply to the turbine-driven AFW pump in Modes 1, 2, and 3, or for the inoperability of the turbine-driven AFW pump if the inoperability occurs in Mode 3 following a refueling outage, if Mode 2 had not been entered. Extending the allowed action time does not create the possibility of a new or different kind of accident from any accident previously evaluated because: 1) the proposed amendment does not represent a change to the system design, 2) the proposed amendment does not alter how equipment is operated or the ability of the system to deliver the required AFW flow, and 3) the proposed amendment does not affect any other safety related equipment. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3: Does the Proposed Amendment Involve a Significant Reduction in a Margin of Safety?

Response: No.

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

The proposed amendment to MPS 2 and 3 TS 3.7.1.2 would allow a 7 day allowed action time for the inoperability of the necessary steam supply to the turbine-driven AFW pump in Modes 1, 2, and 3. Extending the allowed action time does not involve a significant reduction in a margin of safety because: 1) There is a redundant steam supply to the turbine driven AFW pump, 2) the motor-driven AFW pumps are required to be operable when Mode 3 is entered, 3) the motor-driven AFW pumps can provide sufficient flow to remove decay heat and cool the unit to shutdown cooling system entry conditions from power operations, 4) the motor-driven AFW pumps are designed to supply sufficient water to remove decay heat with steam generator pressure at no load conditions to cool the unit to shutdown cooling entry conditions, 5) the proposed change does not change or introduce any new setpoints at which mitigating functions are initiated, 6) no changes to the design parameters of the AFW [system] are being proposed, and 7) no changes in system operation that would impact an established safety margin are being proposed by this change.

The proposed amendment to MPS 2 and 3 TS 3.7.1.2 would also allow a 7 day allowed action time for the inoperability of the turbine-driven AFW pump if the inoperability occurs in Mode 3 following a refueling outage, if Mode 2 had not been entered. Extending the allowed action time does not involve a significant reduction in a margin of safety because: (1) During a return to power operations following a refueling outage, decay heat is at its lowest levels, (2) the motor-driven AFW pumps are required to be operable when Mode 3 is entered, (3) the motor-driven AFW pumps can provide sufficient flow to remove decay heat and cool the unit to shutdown cooling system entry conditions from power operations, (4) the motor-driven AFW pumps are designed to supply sufficient water to remove decay heat with steam generator pressure at no load conditions to cool the unit to shutdown cooling entry conditions, (5) the proposed change does not change or introduce any new setpoints at which mitigating functions are initiated, (6) no changes to the design parameters of the AFW are being proposed, and (7) no changes in system operation that would impact an established safety margin are being proposed by this change

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

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

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

NRC Branch Chief: Darrell J. Roberts.

Duke Energy Corporation, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of amendment request: March 1, 2006.

Description of amendment request: The proposed amendments would revise the Technical Specifications to reconcile the 10 CFR Part 50 and 10 CFR Part 72 criticality requirements for the loading and unloading of dry spent fuel storage canisters in the spent fuel pool (SFP).

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

The applicable accidents are the dropped fuel assembly and drop of the 100 ton spent fuel cask into the SFP. This amendment request does not change the fuel assemblies or any of the Part 50 structures, systems, or components involved in fuel assembly or cask handling or any of the operations involved. Therefore, this amendment request does not affect the probability of an accident previously evaluated.

The proposed change does not increase the consequences of an accident previously evaluated for the following reasons: there is no increase in radiological source terms for the fuel; there is no change to the SFP water level; subcriticality is maintained for normal and accident conditions for the spent fuel storage racks and for cask loading and unloading; and the same boron concentrations that were previously credited for the spent fuel storage racks are assumed in the criticality analysis for cask loading and unloading.

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

Handling of fuel assemblies and the NUHOMS® spent fuel cask have been previously evaluated for Oconee. These activities lead to evaluation of the fuel handling accident (dropped fuel assembly) and drop of the 100 ton spent fuel cask onto spent fuel stored in the Oconee SFP. These elements of the license amendment request (LAR) are not new, and thus do not create the potential for new or different kinds of accidents.

The new element of this LAR is the application of additional criticality controls (*i.e.*, minimum burnup requirements for the fuel assemblies) beyond the 10 CFR 72 controls already in place for the NUHOMS® spent fuel cask. However, application of such criticality controls is not a new activity for Oconee, since similar criticality controls are currently applied to the spent fuel storage racks. Fuel assembly misloading is not a new accident; as discussed in Enclosure 3, Section 6.5, fuel assembly misloading has been considered previously for the NUHOMS® spent fuel cask and for the Oconee spent fuel pool racks. Furthermore, the criticality analysis for cask loading and unloading evaluates the same boron concentrations, moderator temperatures, and misloading scenario as previously evaluated for the spent fuel storage racks. The analysis demonstrates that a criticality accident does not occur under these conditions. It is concluded that the possibility of a criticality accident is not created since application of criticality controls is not new and the analysis demonstrates that criticality does not occur. More generally, this supports the conclusion that the potential for new or different kinds of accidents is not created.

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

This LAR involves the application of additional criticality controls (minimum burnup requirements) to the 10 CFR 72 controls already in place for the NUHOMS® spent fuel cask. The criticality analysis demonstrates subcriticality margins are maintained for normal and accident conditions consistent with 10 CFR 50.68(b) and other NRC guidance. Margins previously established for Oconee's spent fuel storage racks are not altered. Therefore, this LAR does not result in 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: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Branch Chief: Evangelos C. Marinos.

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

Date of amendment request: February 14, 2006.

Description of amendment request: The proposed change will modify the Arkansas Nuclear One, Unit 2 (ANO–2) Technical Specification (TS) Surveillance Requirement 4.6.1.1.a. Specifically, the proposed change will eliminate the requirement to verify containment isolation valves that are maintained locked, sealed, or otherwise secured closed from the monthly position verification. The proposed change will result in reducing radiological exposure to Operations, Health Physics, and Security personnel.

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 accident mitigation features of the plant for previously evaluated accidents are not affected by the proposed change. No changes are proposed to the physical components or to the containment isolation function.

Repositioning of manual containment isolation valves is procedurally controlled and governed by the note that is contained in TS 3.6.3.1, Containment Isolation Valves, which allows opening locked or sealed closed valves on an intermittent basis. The valve position is tracked until it is restored to its original position (locked or deactivated position, as appropriate). While the valve remains open, an individual, in constant communication with the control room staff, is stationed at the valve. If an accident were to occur, the control room staff would direct the individual stationed at the valve to close the valve thereby precluding the release of radioactivity outside containment.

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

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

Response: No.

The proposed change does not change the design, method of operation, or configuration of the plant. The procedural controls that establish the ANO–2 containment valve program controls and include the administrative controls that are associated with the note in TS 3.6.3.1, ensure containment integrity is appropriately established such that no new or different types of accidents are created.

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

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

Response: No.

The proposed change does not change the design basis for any equipment in the plant. The proposed change will exclude verification of the normally locked, sealed, or otherwise secured closed valves, blind flanges, and the deactivated automatic valves; however, the administrative controls applied to these components ensure that containment integrity is maintained.

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

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

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1700 K Street, NW., Washington, DC 20006–3817.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of amendment requests: March 7, 2006.

Description of amendment requests: The proposed amendments would modify the Technical Specifications (TS) of the units by expanding Section 5.5.2, Leakage Monitoring Program, to include the Liquid Waste Disposal System, the Waste Gas System, and the Post-Accident Containment Hydrogen Monitoring System. These systems are currently in the licensee's own leakage monitoring program but are not listed in TS Section 5.5.2. The licensee also proposed to make an editorial change to the section.

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

(1) Does the proposed change involve a significant increase in the probability of occurrence or consequences of any accident previously evaluated?

No. The proposed change would only add the three subject systems to the listing in Section 5.5.2. The licensee is currently performing leakage monitoring of these systems under its own program. Leakage monitoring of these three systems, whether listed in the TS or not, does not have any impact on the initiation of any accident previously analyzed, or on the scenarios and radiological consequences of these accidents. Consequently, 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?

No. The proposed change is purely administrative, and does not involve any change to the design or operation of a system, structure, or component. Consequently, the proposed change leads to no possibility to create 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?

No. The proposed change would not change any assumption, analysis method, calculation model, or acceptance criterion. Accordingly, the proposed change does not involve a significant reduction in a margin of safety.

Based on the NRC staff's analysis, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Attorney for licensee: James M. Petro, Jr., Esquire, One Cook Place, Bridgman, MI 49106.

NRC Branch Chief: L. Raghavan.

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 amendment request: February 16, 2006.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) requirements related to steam generator (SG) tube integrity. The change is consistent with NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.'' The availability of this TS improvement was announced in the **Federal Register** on May 6, 2005 (70 FR 24126) as part of the consolidated line item improvement process (CLIIP).

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on March 2, 2005 (70 FR 10298) as part of the CLIIP. The licensee affirmed the applicability of the model NSHC determination in its application dated February 16, 2006.

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

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

The proposed change requires a SG Program that includes performance criteria that will provide reasonable assurance that the SG tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational leakage.

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

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

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

The consequences of design basis accidents are, in part, functions of the dose equivalent I–131 in the primary coolant and the primary to secondary leakage rates resulting from an accident. Therefore, limits are included in the plant technical specifications for operational leakage and for dose equivalent I–131 in primary coolant to ensure the plant is operated within its analyzed condition. The typical analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gallon per minute with no more than [500 gallons per day or 720 gallons per day] in any one SG, and that the reactor coolant activity levels of dose equivalent I–131 are at the TS values before the accident.

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

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

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

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

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

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

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

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

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

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

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

Attorney for licensee: 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: February 13, 2006.

Description of amendment request: The proposed amendments would make miscellaneous administrative changes by revising Technical Specifications (TS) 3.0 ''Surveillance Requirement (SR) Applicability''; and TS Chapter 5.0, ''Administrative Controls''. The proposed changes will improve TS usability, conformance with the industry standard, NUREG–1431, ''Standard Technical Specifications, Westinghouse Plants'', Revision 3.0 (NUREG–1431) and accuracy.

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 administrative changes to the Prairie Island Nuclear Generating Plant Technical Specifications as follows: Technical Specification 3.0, ''Surveillance Requirement (SR) Applicability'', revise page headers and correct capitalization; and Technical Specification Chapter 5.0, ''Administrative Controls'', correct Topical Report numbers and make format corrections.

The proposed changes are administrative and do not affect plant operation maintenance or testing. These changes do not affect any plant systems which are accident initiators and thus these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

This license amendment request proposes administrative changes to the Prairie Island Nuclear Generating Plant Technical Specifications as follows: Technical Specification 3.0, ''Surveillance Requirement (SR) Applicability'', revise page headers and correct capitalization; and Technical Specification Chapter 5.0, ''Administrative Controls'', correct Topical Report numbers and make format corrections.

The proposed changes are administrative and thus do not create new failure modes or mechanisms and do not generate new accident precursors. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

This license amendment request proposes administrative changes to the Prairie Island Nuclear Generating Plant Technical Specifications as follows: Technical Specification 3.0, ''Surveillance Requirement (SR) Applicability'', revise page headers and correct capitalization; and Technical Specification Chapter 5.0, ''Administrative Controls'', correct Topical Report numbers and make format corrections.

The proposed Technical Specification changes are administrative and do not affect plant operation, maintenance or testing. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment 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 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: February 16, 2006.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) requirements related to steam generator tube integrity. The change is consistent with NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity.'' The availability of this TS improvement was announced in the **Federal Register** on May 6, 2005 (70 FR 24126) as part of the consolidated line item improvement process (CLIIP).

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on March 2, 2005 (70 FR 10298) as part of the CLIIP. The licensee affirmed the applicability of the model NSHC determination in its application dated February 16, 2006.

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

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

The proposed change requires a SG Program that includes performance criteria that will provide reasonable assurance that the SG tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational leakage.

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

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

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

The consequences of design basis accidents are, in part, functions of the dose equivalent 1–131 in the primary coolant and the primary to secondary leakage rates resulting from an accident. Therefore, limits are included in the plant technical specifications for operational leakage and for dose equivalent 1–131 in primary coolant to ensure the plant is operated within its analyzed condition. The typical analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gallon per minute with no more than [500 gallons per day or 720 gallons per day] in any one SG, and that the reactor coolant activity levels of dose equivalent 1–131 are at the TS values before the accident.

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

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

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

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

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

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

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

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

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

For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by the proposed change to the TS. Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

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 Branch Chief: L. Raghavan.

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

Date of amendment request: February 21, 2006.

Description of amendment request: The amendment would revise Technical Specification 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' to exclude portions of the SG tube below the top of the tubesheet in the SGs from periodic tube inspections based on the application of structural analysis and leak rate evaluation results to re-define the primary-to-secondary pressure boundary.

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 previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed change that alters the steam generator inspection criteria does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed change will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident.

Of the applicable accidents previously evaluated, the limiting transients[,] with respect to the proposed [change] to the steam generator tube inspection criteria, are the steam generator tube rupture (SGTR) event and the steam line break (SLB) accident.

During the SGTR event, the required structural integrity margins of the steam generator tubes will be maintained by the presence of the steam generator tubesheet. Steam generator tubes are hydraulically expanded in the tubesheet area. Tube rupture in tubes with cracks in the tubesheet is precluded by the constraint provided by the tubesheet. This constraint results from the hydraulic expansion process, thermal expansion mismatch between the tube and the tubesheet[,] and from the differential pressure between the primary and secondary side [of the steam generator]. Based on this design, the structural margins against burst, discussed in Regulatory Guide (RG) 1.121, ''Bases for Plugging Degraded PWR [Pressurized-Water Reactor] Steam Generator Tubes,'' are maintained for both normal and postulated accident conditions.

The proposed change does not affect other systems, structures, components or operational features. Therefore, the proposed [change results] in no significant increase in the probability [or] the occurrence of a[n] SGTR accident.

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

The probability of an SLB is unaffected by the potential failure of a steam generator tube as this failure is not an initiator for an SLB.

The consequences of an SLB are also not significantly affected by the proposed change. During an SLB accident, the reduction in pressure above the tubesheet on the secondary side of the steam generator creates a uniformly distributed axial (out of plane) load on the tubesheet due to the reactor coolant system pressure on the primary [side] of the tubesheet. The resulting bending action causes contraction of the tube holes below the tubesheet neutral axis, adding to the constraint of the tubes in the tubesheet, thereby further restricting primary-to-secondary leakage.

Primary-to-secondary leakage from tube degradation in the tubesheet area during the limiting accident (*i.e.*, an SLB) is limited by flow restrictions resulting from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of potential crack face opening as compared to free span indications. The primary-to-secondary leak rate from tube degradation in the tubesheet region during postulated SLB accident conditions will be no more than twice that allowed during normal operating conditions when the pressure boundary is relocated [by the amendment] to the lesser of the H* or B* [tubesheet inspection] depths. Since normal operating leakage would be limited to 300 gpd [gallons per day] (0.2 gpm [gallons per minute]) through any one steam generator per TS 3.4.13, ''RCS [Reactor Coolant System] Operational leakage,'' the associated accident condition leak rate, assuming all leakage to be from lower tubesheet indications, would be limited to 150 gpd per steam generator. This value is well within the assumed accident leakage rate of 1.0 gpm discussed in WCGS [(Wolf Creek Generating Station)] Updated Safety Analysis Report, Table 15.1–3, ''Parameters Used in Evaluating the Radiological Consequences of a Main Steam Line Break.'' Therefore, the consequences of an SLB accident remain unaffected.

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 accident from any accident previously evaluated?

Response: No.

The proposed change does not introduce any new equipment, create new failure modes for existing equipment, or create any new limiting single failures. Plant operation will not be altered, and all safety functions will continue to perform as previously assumed in accident analyses. [Excluding portions of the tube below the proposed tubesheet inspection depths does not introduce a new or different kind of accident to the steam generator tube because the required structural margins of the tubes for both normal and accident conditions are maintained.] Therefore, the proposed [change does] not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed [change maintains] the required structural margins of the steam generator tubes for both normal and accident conditions. Nuclear Energy Institute (NEI) 97–06, ''Steam Generator Program Guidelines,'' and RG 1.121, ''Bases for Plugging Degraded PWR Steam Generator Tubes,'' are used as the bases in the development of the tubesheet inspection depth methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC for meeting General Design Criteria (GDC) 14, ''Reactor coolant pressure boundary,'' GDC 15, ''Reactor coolant system design,'' GDC 31, ''Fracture prevention of reactor coolant pressure boundary,'' and GDC 32, ''Inspection of reactor coolant pressure boundary,'' by reducing the probability and consequences of a[n] SGTR. RG 1.121 concludes that by determining the limiting safe conditions for tube wall degradation[,] the probability and consequence of a[n] SGTR are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code.

For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, Westinghouse letter LTR-CDME–05–209-P, ''Steam Generator Tube Alternate Repair Criteria for the Portion of the Tube Within the Tubesheet at the Wolf Creek Generating Station,'' [provided in the application,] defines a length of degradation-free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces, with applicable safety factors applied. Application of the limited tubesheet inspection depth criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for large margins between calculated and actual leakage values in the proposed limited tubesheet inspection depth criteria.

Therefore, the proposed [change does not] involve a significant reduction in any 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: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, DC 20037.

NRC Branch Chief: David Terao.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: February 25, 2005.

Brief description of amendment: The amendment deleted the reporting requirement in the Facility Operating License (FOL) related to reporting violations of other requirements in the operating license.

Date of issuance: February 24, 2006.

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

Amendment No.: 172.

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

*Date of initial notice in **Federal Register***: April 26, 2005 (70 FR 21450).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 25, 2005.

Brief description of amendment: The amendment revised the Technical Specifications (TSs) to exclude the containment purge valve leakage rates from the summation of secondary containment bypass leakage rates.

Date of issuance: March 21, 2006.

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

Amendment No.: 173.

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

*Date of initial notice in **Federal Register***: April 26, 2005 (70 FR 21451).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 13, 2005, as supplemented on November 29, 2005, and January 20 and February 13, 2006.

Brief description of amendments: The amendments revised Technical Specification (TS) 1.1, ''Definitions,'' TS 3.4.13, ''RCS [reactor coolant system] Operational Leakage,'' TS 5.5.9, ''Steam Generator Tube Surveillance Program,'' and TS 5.6.9, ''Steam Generator [SG] Tube Inspection Report,'' and add a new specification (TS 3.4.18) for SG Tube Integrity. The changes are consistent with TS Task Force (TSTF) Change TSTF–449, Revision 4, ''Steam Generator Tube Integrity.''

Date of issuance: March 9, 2006.

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

Amendment Nos.: 278 and 255.

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

*Date of initial notice in **Federal Register***: December 6, 2005 (70 FR 72669).

The November 29, 2005, and January 20 and February 13, 2006, supplements 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 these amendments is contained in a Safety Evaluation dated March 9, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 25, 2005, as supplemented by letter dated January 23, 2006.

Brief description of amendment: The amendment revises the Technical Specification limit on pressurizer water level in Mode 3 (hot standby).

Date of issuance: March 22, 2006.

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

Amendment No.: 246.

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

*Date of initial notice in **Federal Register***: June 21, 2005 (70 FR 35736).

The January 23, 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 March 22, 2006.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: October 3, 2005.

Brief description of amendment: The amendment revises Technical Specification Surveillance Requirements to reflect changes to the Emergency Core Cooling System throttle valves. The amendment adds the modified throttle valves to the surveillance, removes existing throttle valves that are now locked closed from the surveillance, and adds existing valves to the surveillance that are used in a throttle position when open.

Date of issuance: March 23, 2006.

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

Amendment No.: 230.

Facility Operating License No. DPR–64: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 6, 2005 (70 FR 72670).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 23, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 25, 2005.

Brief description of amendment: The amendments delete the sections of the Facility Operating Licenses that require reporting of violations of the requirements in Section 2.C of the Facility Operating Licenses.

Date of issuance: March 13, 2006.

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

Amendment Nos.: 146, 146, 139 and 139.

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

*Date of initial notice in **Federal Register:*** April 26, 2005 (70 FR 21456).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 13, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 4, 2005, as supplemented by letter dated January 13, 2006.

Brief description of amendment: The amendments revised Technical Specification 3.3.8.1, "Loss of Power (LOP) Instrumentation," and also revised the Updated Final Safety Analysis Report to implement use of automatic load tap changers on transformers that provide offsite power to DNPS, Units 2 and 3.

Date of issuance: March 17, 2006.

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

Amendment Nos.: 219/210.

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

*Date of initial notice in **Federal Register:*** November 8, 2005 (70 FR 67747).

The January 13, 2006 supplement, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

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

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket No. 50–352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of application for amendment: January 10, 2005.

Brief description of amendment: The amendment removed the license conditions concerning the emergency core cooling system pump suction strainers from Appendix C of Facility Operating License No. NPF–39.

Date of issuance: March 6, 2006.

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

Amendment No.: 184.

Facility Operating License No. NPF–39.

This amendment revised the License.

*Date of initial notice in **Federal Register:*** January 3, 2006 (71 FR 149).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 5, 2006.

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: September 29, 2005.

Description of amendment request: The proposed amendment would revise the Seabrook Station, Unit No. 1 Technical Specifications (TSs) to permit a one-time, 6-month addition to the currently approved 5-year extension to the 10-year test interval for the containment integrated leak rate test.

Date of issuance: March 24, 2006.

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

Amendment No.: 108.

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

Date of initial notice in **Federal Register**: November 8, 2005 (70 FR 67748).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 26, 2005.

Brief description of amendment: The amendment revises Technical Specification 5.6.5.b, ''Core Operating Limits Report,'' to use a revised fuel assembly growth model for Palisades as described in Topical Report BAW–2489P, ''Revised Fuel Assembly Growth Correlation for Palisades,'' Revision 0.

Date of issuance: March 27, 2006.

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

Amendment No.: 222.

Facility Operating License No. DPR–20. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: (70 FR 29797).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 1, 2005, as supplemented on February 17, 2006.

Brief description of amendment: The amendment changes the SSES 1 Technical Specifications (TSs) by revising the Unit 1 Cycle 15 Minimum Critical Power Ratio Safety Limit for single-loop operation in TS 2.1.1.2 and the references listed in TS 5.6.5.b.

Date of issuance: March 20, 2006.

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

Amendment No.: 231.

Facility Operating License No. NPF–14: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 17, 2006 (71 FR 2595).

The supplement dated February 17, 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 20, 2006.

No significant hazards consideration comments received: No.

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

Date of amendment request: October 6, 2004, as supplemented by letters dated September 16 and November 22, 2005.

Brief description of amendments: The amendments revised the Technical Specification 3.8.1, ''AC Sources—Operating,'' to remove mode restrictions on surveillance requirements.

Date of issuance: March 15, 2006.

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

Amendment Nos.: 124 and 124.

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

Date of initial notice in **Federal Register**: March 15, 2005 (70 FR 12751).

The supplements dated September 16 and November 22, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 15, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 26, 2005.

Brief description of amendment: The amendment revised Required Action D.1, in Technical Specification (TS) 3.6.6, ''Containment Spray and Cooling Systems,'' to require plant shutdown if both containment cooling trains are out of service, which is more conservative than the previous requirement that allowed 72 hours to restore one of the inoperable trains. There are also changes to other required actions in TS 3.6.6 to reflect the revision to Required Action D.1.

Date of issuance: March 28, 2006.

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

Amendment No.: 171.

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

Date of initial notice in **Federal Register:**January 17, 2006 (71 FR 2597).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 28, 2006.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 3rd day of April 2006.

For the Nuclear Regulatory Commission.

Catherine Haney,

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

[FR Doc. E6–5086 Filed 4–10–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Notice of Opportunity To Comment on Model Safety Evaluation on Technical Specification Improvement Regarding Revision to the Completion Time in STS 3.6.6A, ''Containment Spray and Cooling Systems'' for Combustion Engineering Pressurized Water Reactors Using the Consolidated Line Item Improvement Process

AGENCY: Nuclear Regulatory Commission.

ACTION: Request for comment.

SUMMARY: Notice is hereby given that the staff of the U.S. Nuclear Regulatory Commission (NRC) has prepared a model license amendment request (LAR), model safety evaluation (SE), and model proposed no significant hazards consideration (NSHC) determination related to changes to the completion times (CT) in Standard Technical Specification (STS) 3.6.6A, ''Containment Spray and Cooling Systems.'' The proposed changes would revise STS 3.6.6A by extending the CT for one containment spray system (CSS) train inoperable from 72 hours to seven days, and add a Condition describing required Actions and CT when one CSS and one containment cooling system (CCS) are inoperable. These changes are based on analyses provided in a joint applications report submitted by the Combustion Engineering Owner's Group (CEOG). The CEOG participants in the Technical Specifications Task Force (TSTF) proposed this change to the STS in Change Traveler No. TSTF–409, Revision 2.

The purpose of these models is to permit the NRC to efficiently process amendments to incorporate these changes into plant-specific STS for Combustion Engineering pressurized water reactors (PWRs). Licensees of nuclear power reactors to which the models apply can request amendments conforming to the models. In such a