For the Nuclear Regulatory Commission.

Dated this 11th day of July 2006 at Rockville, Maryland.

Margaret M. Doane,

Deputy Director, Office of International Programs.

[FR Doc. E6–12369 Filed 7–31–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–151]

Notice and Solicitation of Comments Concerning Proposed Action To Decommission University of Illinois at Urbana-Champaign Nuclear Reactor Laboratory

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has received an application from the University of Illinois at Urbana-Champaign dated March 28, 2006, for a license amendment approving its proposed decommissioning plan for the Nuclear Reactor Laboratory (Facility License No. R–115) located in Urbana, Illinois.

In accordance with 10 CFR 20.1405, the Commission is providing notice and soliciting comments from local and State governments in the vicinity of the site and any Indian Nation or other indigenous people that have treaty or statutory rights that could be affected by the decommissioning. This notice and solicitation of comments is published pursuant to 10 CFR 20.1405, which provides for publication in the **Federal Register** and in a forum, such as local newspapers, letters to State or local organizations, or other appropriate forum, that is readily accessible to individuals in the vicinity of the site.

Comments should be provided within 60 days of the date of this notice to Alexander Adams, Jr., Senior Project Manager, U.S. Nuclear Regulatory Commission, Research and Test Reactors Branch, MS O–12–G–15, Washington, DC 20555.

Further, in accordance with 10 CFR 50.82(b)(5), notice is also provided to interested persons of the Commission's intent to approve the plan by amendment, subject to such conditions and limitations as it deems appropriate and necessary, if the plan demonstrates that decommissioning will be performed in accordance with the regulations and will not be inimical to the common defense and security or to the health and safety of the public.

A copy of the application (Accession Number ML060900623) is available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at (the Public Electronic Reading Room) *http://www.nrc.gov/reading-rm/adams.html.*

Dated at Rockville, Maryland, this 25th day of July 2006.

For the Nuclear Regulatory Commission.

Brian E. Thomas,

Branch Chief, Research and Test Reactors Branch, Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation.

[FR Doc. E6–12371 Filed 7–31–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Federal Register Notice

AGENCY HOLDING THE MEETINGS: Nuclear Regulatory Commission.

DATE: Weeks of July 31, August 7, 14, 21, 28, September 4, 2006.

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

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of July 31, 2006

There are no meetings scheduled for the Week of July 31, 2006.

Week of August 7, 2006—Tentative

There are no meetings scheduled for the Week of August 7, 2006.

Week of August 14, 2006—Tentative

There are no meetings scheduled for the Week of August 14, 2006.

Week of August 21, 2006—Tentative

There are no meetings scheduled for the Week of August 21, 2006.

Week of August 28, 2006—Tentative

There are no meetings scheduled for the Week of August 28, 2006.

Week of September 4, 2006—Tentative

There are no meetings scheduled for the Week of September 4, 2006.

The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings, call (recording)—(301) 415–1292. Contact person for more information: Michelle Schroll, (301) 415–1662.

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

The NRC provides reasonable accommodation to individuals with disabilities where appropriate. If you need a reasonable accommodation to participate in these public meetings, or need this meeting notice or the transcript or other information from the public meetings in another format (*e.g.* braille, large print), please notify the NRC's Disability Program Coordinator, Deborah Chan, at 301–415–7041, TDD: 301–415–2100, or by e-mail at *DLC@nrc.gov*. Determinations on requests for reasonable accommodation will be made on a case-by-case basis.

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

Dated: July 27, 2006.

Sandy Joosten,

Office of the Secretary.

[FR Doc. 06–6628 Filed 7–28–06; 9:47 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from July 7, 2006 to July 19, 2006. The last biweekly notice was published on July 18, 2006 (71 FR 40742).

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** 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.*

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

Date of amendments request: September 29, 2005, as supplemented by letter dated July 5, 2006.

Description of amendments request: The amendments revised the Physical Security Plan to clarify the description of the owner controlled area vehicle checkpoint.

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, which will clarify the description of a security feature of the Owner Controlled Area (OCA) Checkpoint, does not reduce the ability of the Security organization to prevent radiological sabotage and, therefore, does not increase the probability or consequences of a radiological release previously evaluated. The proposed Security Plan changes will not affect any important to safety systems or components, their mode of operation or operating strategies. The proposed Security Plan changes have no affect on accident initiators or mitigation. Therefore, the proposed amendment to the Security Plan will 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 amendment to clarify the description of a security feature of the OCA Checkpoint does not affect the operation of systems important to safety. The Security Plan amendment does not affect any of the parameters or conditions that could contribute to the initiation of any accident. No new accident scenarios are created as a result of the proposed Security Plan changes. In addition, the design functions of equipment important to safety are not altered as a result of the proposed Security Plan changes. Therefore, the proposed Security Plan changes will not create the possibility of a new or different accident from any previously evaluated.

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

Response: No.

The proposed Security Plan changes will not affect any important to safety systems or components, their mode of operation, or operating strategies. The proposed Security Plan changes have no affect on accident initiators or mitigation. The proposed amendment to the Security Plan does not reduce the effectiveness of any security/safeguards measures currently in place. Therefore, the proposed Security Plan changes will not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Janet S. Mueller, Director, Law Department, Arizona Public Service Company, P.O. Box 52034, Mail Station 8695, Phoenix, Arizona 85072–2034.

NRC Branch Chief: David Terao.

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

Date of amendment request: June 28, 2006.

Description of amendment request: The proposed amendment changed Kewaunee Power Station (KPS) Technical Specifications 3.3.b.3.B and 3.3.b.4.A to increase the minimum required boron concentration in the refueling water storage tank (RWST) from 2400 parts per million (ppm) to 2500 ppm.

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

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

Response: No.

Increasing the minimum required boron concentration in the RWST does not add, delete, or modify any KPS systems, structures, or components (SSCs). The RWST and its contents are not accident initiators. Rather, they are designed for accident mitigation. The effects of an increase in the minimum RWST boron concentration from 2400 ppm to 2500 ppm are bounded by existing evaluations and determined to be acceptable. Thus, the proposed increase in minimum RWST boron concentration has no adverse effect on the ability of the plant to mitigate the effects of design basis accidents.

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

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

Response: No.

Increasing the minimum required boron concentration in the RWST does not change the design function of the RWST or the SSCs designed to deliver borated water in the RWST to the [reactor] core. Increasing the minimum required boron concentration in the RWST does not create any credible new failure mechanisms or malfunctions for plant equipment or the nuclear fuel. The safety function of the borated water in the RWST is not being changed.

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

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

Response: No.

An evaluation has been performed showing that maintaining RWST boron concentration above 2500 ppm continues to assure acceptable results for design basis accident analyses [ ] considering the reactivity of the core. Increasing the minimum boron concentration in the RWST from 2400 ppm to 2500 ppm increases the margin of safety in the KPS safety analyses, since additional post-accident negative reactivity will be available to the core. This additional negative reactivity more than compensates for the additional reactivity in the core due to the unanticipated prolonged shutdown periods in Cycle 27. Additionally, the proposed new minimum boron concentration of 2500 ppm is within the range required by current safety analyses (i.e., 2400 ppm to 2625 ppm), and well below the currently acceptable maximum boron concentration of 2625 ppm.

The proposed amendment does not result in altering or exceeding a design basis or safety limit for the plant. All current fuel design criteria will continue to be satisfied, and the safety analyses of record (except for the postLOCA sump boron concentration), including evaluations of the radiological consequences of design basis accidents, will remain applicable.

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

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

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

NRC Branch Chief: L. Raghavan.

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 amendment request: May 31, 2006.

Description of amendment request: The proposed amendment revised the Technical Specification (TS) requirements related to steam generator (SG) tube integrity. Specifically, it would revise the TS definition of LEAKAGE; TS 3.4.13, ''Reactor Coolant System (RCS) Operational Leakage;'' TS 5.5.7 (Indian Point Unit 2) and TS 5.5.8 (Indian Point Unit 3), ''Steam Generator (SG) Program;'' TS 5.6.7 (Indian Point Unit 2) and TS 5.6.8 (Indian Point Unit 3), ''SG Tube Inspection Report;'' and would create new TS 3.4.17, ''SG Tube Integrity.''

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF 449, Revision 4. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on March 2, 2005 (70 FR 10298), on possible amendments concerning TSTF–449, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process (CLIIP). The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on May 6, 2005 (70 FR 24126). The licensee affirmed the applicability of the following NSHC determination in its application dated May 31, 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, 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 steam generator tube rupture (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 Nuclear Energy Institute (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 a main steam line break (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.

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

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

NRC Branch Chief: Richard J. Laufer.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant, Units 3 and 4, Miami-Dade County, Florida

Date of amendment request: April 27, 2006.

Description of amendment request: The proposed amendments revised the Technical Specifications (TSs) relating to Steam Generator (SG) inspection. Specifically, TS 3/4.4.5, Surveillance Requirements, and TS 3/4.4.6, Reactor Coolant System Leakage, would be modified to clearly delineate the scope of the inservice inspections required in the tube sheet regions of the SGs.

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

Of the various accidents previously evaluated, the proposed changes only affect the SG tube rupture (SGTR) event evaluation and the postulated steam line break [SLB] accident evaluation. Loss-of-coolant accident (LOCA) conditions cause a compressive axial load to act on the tube. Therefore, since the LOCA tends to force the tube into the tubesheet rather than pull it out, it is not a factor in this amendment request. Another faulted load consideration is a safe shutdown earthquake (SSE); however, the seismic analysis of Series 44F SGs has shown that axial loading of the tubes is negligible during a SSE.

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

The limited inspection length of 17 inches supplies the necessary resistive force to preclude pullout loads under both normal operating and accident conditions. The contact pressure results from the hydraulic expansion process, thermal expansion mismatch between the tube and tubesheet and from the differential pressure between the primary and secondary side. The proposed changes do not affect other systems, structures, components or operational features. Therefore, the proposed change results in no significant increase in the probability of the occurrence of a SGTR event.

The consequences of an SGTR event are affected by the primary-to-secondary leakage flow during the event. Primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed change since the tubesheet enhances the tube integrity in the region of the hydraulic expansion by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and collapse is strengthened by the tubesheet in that region. At normal operating pressures, leakage from primary water stress corrosion cracking (PWSCC) below 17 inches from the top of the tubesheet 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 probability of a SLB is unaffected by the potential failure of a SG tube as the failure of a tube is not an initiator for a SLB event. SLB leakage is limited by leakage flow restrictions resulting from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of crack face opening compared to free span indications. The leak rate during postulated accident conditions would be expected to be less than twice that during normal operation for indications near the bottom of the tubesheet (including indications in the tube end welds) based on the observation that while the driving pressure increases by about a factor of two, the flow resistance increases with an increase in the tube-to-tubesheet contact. While such a decrease is rationally expected, the postulated accident leak rate is bounded by twice the normal operating leak rate if the increase in contact pressure is ignored. Since normal operating leakage is limited to less than 150 gpd, the attendant accident condition leak rate, assuming all leakage to be from lower tubesheet indications, would be bounded by 300 gpd. This value is less than the 500 gpd leak rate assumed during a postulated SLB in the Turkey Point Units 3 and 4 Updated Final Safety Analysis Report (UFSAR).

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

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

The proposed changes do not introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Tube bundle integrity is expected to be maintained for all plant conditions upon implementation of the limited tubesheet inspection depth methodology. The proposed changes do not introduce any new equipment or any change to existing equipment. No new effects on existing equipment are created nor are any new malfunctions introduced.

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

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

The proposed changes maintain the required structural margins of the SG tubes for both normal and accident conditions. NEI [Nuclear Energy Institute] 97–06, Rev. 2 and RG 1.121 are used as the basis in the development of the limited tubesheet inspection depth methodology for determining that SG tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC staff for meeting General Design Criteria 14, 15, 31, and 32 by reducing the probability and consequences of an SGTR. RG 1.121 concludes that by determining the limiting safe conditions of tube wall degradation beyond which tubes with unacceptable cracking, as established by inservice inspection, should be removed from service or repaired, the probability and consequences of a SGTR are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the ASME [American Society of Mechanical Engineers] Code.

For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, WCAP [Westinghouse Commercial Atomic Power] —16506–P 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.

Plugging of the SG tubes reduces the reactor coolant flow margin for core cooling. Implementation of the 17 inch inspection length at Turkey Point Units 3 and 4 will result in maintaining the margin of flow that may have otherwise been reduced by tube plugging.

Based on the above, it is concluded that the proposed changes do not result in any reduction of margin with respect to plant safety as defined in the UFSAR or Bases of the plant Technical Specifications.

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

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

NRC Branch Chief: Michael L. Marshall, Jr.

FPL Energy Duane Arnold, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: November 14, 2005.

Description of amendment request: The proposed amendment revised the table of Primary Containment Isolation Instrumentation to eliminate the trip generated by the main steamline radiation monitors.

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

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

Response: No.

The proposed change deletes the Main Steamline Radiation Monitor (MSLRM) trip function from TS [technical specification]. The MSLRM is not an initiator of any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The consequences of an accident previously evaluated, specifically the Control Rod Drop Accident (CRDA), have been evaluated consistent with the DAEC [Duane Arnold Energy Center] licensing basis utilizing the Alternative Source Term (10 CFR 50.67). As demonstrated by the dose calculations, the consequences of the accident are within the regulatory acceptance criterion. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

No new or different accidents result from utilizing the proposed change. The changes do not involve a change in the methods governing normal plant operation. The equipment proposed to be removed from the plant, the MSLRM, is only credited in the CRDA analysis and no other event in the safety analysis. The proposed changes are consistent with the revised safety analysis assumptions for a CRDA included in this application.

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

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

Response: No.

The proposed change deletes the requirement for the MSLRM isolation function. Analyses performed consistent with the DAEC licensing basis, demonstrate that the removal of this isolation will not cause a significant reduction in the margin of safety, as the resulting offsite dose consequences are being maintained within regulatory limits.

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

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

Attorney for licensee: Mr. R.E. Helfrich, Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408–0420.

NRC Branch Chief: L. Raghavan.

FPL Energy Duane Arnold, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: December 22, 2005.

Description of amendment request: The proposed amendment revised the reactor-pressure vessel material surveillance program described within the Duane Arnold Energy Center (DAEC) Updated Final Safety Analysis Report from a plant-specific program to the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP).

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

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

Response: No.

The proposed change implements an integrated surveillance program that has been evaluated by the NRC [Nuclear Regulatory Commission] staff as meeting the requirements of paragraph III.C of Appendix H to 10 CFR 50. Consequently, the proposed change does not significantly increase the probability of any accident previously evaluated. The proposed change provides the same assurance of RPV [reactor pressure vessel] integrity. As a result, the consequences of any accident previously evaluated are not significantly increased.

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

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

Response: No.

The proposed change revises the DAEC licensing bases to reflect participation in the BWRVIP ISP. The ISP was approved by the NRC staff as an acceptable material surveillance program which complies with 10 CFR 50, Appendix H. The proposed change maintains an equivalent level of RPV material surveillance and does not introduce any new accident initiators. The proposed change will not impact the manner in which the plant is designed or operated. This change will not affect the reactor pressure vessel, as no physical changes are involved. The proposed change will not cause the reactor pressure vessel or interfacing systems to be operated outside of any design or testing limits. Furthermore, the proposed changes will not alter any assumptions previously made in evaluating the radiological consequences of any accident.

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

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

Response: No.

The proposed change has been evaluated as providing an acceptable alternative to the plant-specific RPV material surveillance program that meets the requirements of the regulations for RPV material surveillance. The material surveillance program requirements contained in 10 CFR 50, Appendix H provide assurance that adequate margins of safety exist for the reactor coolant system against nonductile or rapidly propagating failures during normal operation, anticipated operational occurrences, and system hydrostatic tests.

The BWRVIP ISP has been approved by the NRC staff as an acceptable material surveillance program which complies with I0 CFR 50, Appendix H. The ISP will provide the material surveillance data which will ensure that the safety margins required by NRC regulations are maintained for the DAEC reactor coolant system.

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

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

Attorney for licensee: Mr. R. E. Helfrich, Florida Power & Light Company, P. O. Box 14000, Juno Beach, FL 33408–0420.

NRC Branch Chief: L. Raghavan.

FPL Energy Duane Arnold, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: April 28, 2006.

Description of amendment request: The proposed amendment modified technical specifications (TSs) requirements for inoperable snubbers by adding Limiting Condition for Operation (LCO) 3.0.8. The changes are consistent with Nuclear Regulatory Commission approved Industry/Technical Specification Task Force (TSTF) standard TS change TSTF–372, Revision 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 May 4, 2005 (70 FR 23252). The licensee affirmed the applicability of the model NSHC determination in its application dated April 28, 2006.

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

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

Response: No.

The proposed change allows a delay time before declaring supported TS systems inoperable when the associated snubber(s) cannot perform its required safety function. Entrance into Actions or delaying entrance into Actions is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The consequences of an accident while relying on the delay time allowed before declaring a TS supported system inoperable and taking its Conditions and Required Actions are no different than the consequences of an accident under the same plant conditions while relying on the existing TS supported system Conditions and Required Actions. Therefore, the consequences of an accident previously evaluated are not significantly increased by this change. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change allows a delay time before declaring supported TS systems inoperable when the associated snubber(s) cannot perform its required safety function. 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 accident previously evaluated.

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

Response: No.

The proposed change allows a delay time before declaring supported TS systems inoperable when the associated snubber(s) cannot perform its required safety function. The proposed change restores an allowance in the pre-ISTS conversion TS that was unintentionally eliminated by the conversion. The pre-ISTS TS were considered to provide an adequate margin of safety for plant operation, as does the post-ISTS conversion TS. Therefore, this 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: Mr. R.E. Helfrich, Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408–0420.

NRC Branch Chief: L. Raghavan.

Indiana Michigan Power Company, Docket No. 50–315, Donald C. Cook Nuclear Plant, Unit 1, Berrien County, Michigan

Date of amendment request: April 10, 2006.

Description of amendment request: The proposed amendment revised Surveillance Requirement 3.8.1.11 of the Donald C. Cook Technical Specifications, raising the emergency diesel generator full load rejection voltage test limit from 5000 volts to 5350 volts.

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

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

Response: No.

Probability of Occurrence of an Accident Previously Evaluated.

The proposed change is an increase in the Technical Specification (TS) Surveillance Requirement (SR) limit on maximum voltage following an emergency diesel generator (DG) full load rejection. The DGs' safety function is solely mitigative and is not needed unless there is a loss of offsite power. The DGs do not affect any accident initiators or precursors of any accident previously evaluated. The proposed increase in the TS SR limit does not affect the DGs' interaction with any system whose failure or malfunction can initiate an accident. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased.

Consequences of an Accident Previously Evaluated.

The DG safety function is to provide power to safety related components needed to mitigate the consequences of an accident following a loss of offsite power. The purpose of the TS SR voltage limit is to assure DG damage protection following a full load rejection. The technical analysis performed to support this proposed amendment has demonstrated that the DGs can withstand voltages above the new proposed limit without a loss of protection. The proposed higher limit will continue to provide assurance that the DG is protected, and the safety function of the DG will be unaffected by the proposed change. Therefore, the consequences of an accident previously evaluated will not be significantly increased.

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

Response: No.

There are no new DG failure modes created and the DGs are not an initiator of any new or different kind of accident. The proposed increase in the TS SR limit does not affect the interaction of the DGs with any system whose failure or malfunction can initiate an accident. 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 margins of safety applicable to the proposed change are those associated with the ability of the DGs to perform their safety function. The technical analysis performed to support this amendment demonstrates that this ability will be unaffected. The increase in the TS SR limit will not affect this ability. Therefore, the proposed change does not involve a significant reduction in margin of safety.

The NRC staff evaluated the licensee's analysis, and based on this evaluation, the NRC staff proposes to determine that the requested amendment does not involve a significant hazards consideration.

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

NRC Branch Chief: L. Raghavan.

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

Date of amendment request: June 16, 2006.

Description of amendment request: The proposed amendment revised Technical Specification (TS) 3.10.1, ''Inservice Leak and Hydrostatic Testing Operation,'' to extend the scope to include provisions for temperature increases above 212 °F as a consequence of inservice leak or hydrostatic testing, and as a consequence of control rod scram time testing initiated in conjunction with the inservice leak test or hydrostatic test, when initial test conditions are below 212 °F.

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

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

Response: No.

Current TS LCO [Limiting Condition for Operation] 3.10.1 allows average RCS [reactor coolant system] temperature to exceed 212 °F when required during the conduct of hydrostatic and inservice leak tests without requiring entry into plant operating Mode 3, Hot Shutdown. Extending this allowance to testing in which average RCS temperature exceeds 212 °F as a consequence of maintaining pressure and to the performance of scram time testing that is initiated in conjunction with the hydrostatic and inservice leak tests will not impact any accident initiator. Thus, the proposed change does not affect the probability of any accident.

The proposed changes do not involve any modification of equipment used to mitigate accidents, and do not impact any system used in the mitigation of design basis accidents. The proposed changes do not involve modified operation of equipment or [a] system used to mitigate accidents. Thus, the proposed changes do not affect the consequences of an accident.

Based on the above, NPPD concludes that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed TS revisions to TS LCO 3.10.1 do not involve physical modification of the plant or a change in plant operation. The proposed TS revisions do not revise or eliminate any existing requirements, and do not impose any additional requirements. The proposed changes do not alter assumptions made in the safety analysis, and are consistent with the safety analysis assumptions and current plant operating practice. Allowing the performance of control rod scram time testing, while in plant operating Mode 4 with average RCS temperature greater than 212 °F, does not create the possibility of a different kind of accident.

Based on the above NPPD[,] concludes that these 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.

The proposed changes do not impact the design or operation of the Reactor Protection System or the Emergency Core Cooling System. Allowing completion of scram time testing that was initiated in conjunction with inservice leak or hydrostatic testing prior to reactor criticality and startup will eliminate the need for unnecessary plant maneuvers to control reactor temperature and pressure, thereby resulting in enhanced safe operation.

Based on the above, NPPD concludes that these proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mr. John C. McClure, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Branch Chief: David Terao.

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

Date of amendment request: January 18, 2006.

Description of amendment request: The proposed amendment deleted the reference to the hydrogen monitors in Technical Specification (TS) 3.6.11, ''Accident Monitoring Instrumentation'' consistent with the NRC-approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–447, ''Elimination of Hydrogen Recombiners and Change to Hydrogen and Oxygen Monitors.''

The NRC staff issued a notice of availability of ''Model Application Concerning Technical Specification Improvement To Eliminate Hydrogen Recombiner Requirement, and Relax the Hydrogen and Oxygen Monitor Requirements for Light Water Reactors Using the Consolidated Line Item Improvement Process (CLIIP)'', in the **Federal Register** on September 25, 2003 (68 FR 55416). The notice included a model safety evaluation (SE), a model no significant hazards consideration (NSHC) determination, and a model application.

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, by confirming the applicability of the model NSHC determination to NMP–1 and incorporating it by reference in its application. The model NSHC determination 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 revised 10 CFR 50.44 no longer defines a design-basis loss-of-coolant accident (LOCA) hydrogen release, and eliminates requirements for hydrogen control systems to mitigate such a release. The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. In addition, these systems were ineffective at mitigating hydrogen releases from risk-significant accident sequences that could threaten containment integrity.

With the elimination of the design-basis LOCA hydrogen release, hydrogen [and oxygen] monitors are no longer required to mitigate design-basis accidents and, therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. RG [Regulatory Guide] 1.97 Category 1, is intended for key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. The hydrogen [and oxygen] monitors no longer meet the definition of Category 1 in RG 1.97. As part of the rulemaking to revise 10 CFR 50.44 the Commission found that Category 3, as defined in RG 1.97, is an appropriate categorization for the hydrogen monitors because the monitors are required to diagnose the course of beyond design-basis accidents. [Also, as part of the rulemaking to revise 10 CFR 50.44, the Commission found that Category 2, as defined in RG 1.97, is an appropriate categorization for the oxygen monitors, because the monitors are required to verify the status of the inert containment.]

The regulatory requirements for the hydrogen [and oxygen] monitors can be relaxed without degrading the plant's, emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. Classification of the hydrogen monitors as Category 3, [classification of the oxygen monitors as Category 2] and removal of the hydrogen [and oxygen] monitors from TS will not prevent an accident management strategy through the use of the SAMGs [severe accident management guidelines], the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of the hydrogen recombiner requirements and relaxation of the hydrogen [and oxygen] monitor requirements, including removal of these requirements from TS, does not involve a significant increase in the probability or the consequences of any accident previously evaluated.

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

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen [and oxygen] monitor requirements, including removal of these requirements from TS, will not result in any failure mode not previously analyzed. The hydrogen recombiner and hydrogen [and oxygen] monitor equipment was intended to mitigate a design-basis hydrogen release. The hydrogen recombiner and hydrogen [and oxygen] monitor equipment are not considered accident precursors, nor does their existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

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

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen [and oxygen] monitor requirements, including removal of these requirements from TS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety.

The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage.

Category 3 hydrogen monitors are adequate to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The intent of the requirements established as a result of the TMI [Three Mile Island], Unit 2 accident can be adequately met without reliance on safety-related hydrogen monitors.

[Category 2 oxygen monitors are adequate to verify the status of an inerted containment.]

Therefore, this change does not involve a significant reduction in [a] margin of safety. [The intent of the requirements established as a result of the TMI, Unit 2 accident can be adequately met without reliance on safety-related oxygen monitors.]

Removal of hydrogen [and oxygen] monitoring from TS will not result in a significant reduction in their functionality, reliability, and availability.

The NRC staff has reviewed the model NSHC determination and its applicability to NMP–1. Based on this review, 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 Branch Chief: Richard J. Laufer.

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

Date of amendment request: June 7, 2006.

Description of amendment request: The amendment deleted Required Action D.1.2 in Technical Specification (TS) 3.7.10, ''Control Room Emergency Ventilation System (CREVS),'' and Required Action C.1.2 in TS 3.7.11, ''Control Room Air Conditioning System (CRACS).'' These required actions are for the condition where the required actions and completion time (CT) of TS 3.7.10 Condition A (one CREVS train inoperable) and TS 3.7.11 Condition A (one CRACS train inoperable) are not met in Modes 5 or 6, or during movement of irradiated fuel assemblies. The deleted required actions, and associated CTs, are to verify the operable CREVS (or CRACS) train is capable of being powered by an emergency power source.

The amendment would also delete the phrase ''in MODES 1, 2, 3, or 4'' from Condition A (one emergency exhaust system (EES) train inoperable) of TS 3.7.13, ''Emergency Exhaust System (EES),'' and revise Condition D to state the following: ''Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the fuel building.''

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.

Incorporation of a 7-day Completion Time for restoring an inoperable EES train during shutdown conditions (*i.e.*, during movement of irradiated fuel assemblies in the fuel building) and the deletion of Required Actions for verifying the availability of an emergency power source when a CREVS/ CRACS train is inoperable during the same [shutdown] conditions, are operational provisions that have no impact on the frequency of occurrence of the event for which the EES, CREVS and CRACS are designed to mitigate, *i.e.*, a fuel handling accident (FHA) in the fuel building. These systems, (*i.e.*, their failure)[,] have no bearing on the occurrence of a fuel handling accident as the systems themselves are not associated with any of the potential initiating sequences, mechanisms or occurrences—such as failure of a lifting device or crane [lifting a fuel assembly], or an operator error—that could cause an FHA. Since these systems are designed only to respond to an FHA as accident mitigators after the accident has occurred, and they have no bearing on the occurrence of such an event themselves, the proposed changes to the CREVS, CRACS and EES Technical Specifications have no impact on the probability of occurrence of an FHA. On this basis, the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

With regard to [the] consequences of previously evaluated accidents (*i.e.*, an FHA), the proposed changes involve no design or physical changes to the EES or any other equipment required for accident mitigation.

With respect to deleting the noted Required Actions (for verifying that the operable CREVS/CRACS train is capable of being powered from an emergency power source when on CREVS/CRACS train is inoperable), such a change does not change the Limiting Condition for Operation (LCO) requirement for both CREVS/CRACS trains to be operable, nor to the LCO requirements of the TS requirements pertaining to electrical power sources/support for shutdown conditions. The change to the Required Actions would thus not be expected to have a significant impact on the availability of the CREVS and CRACS. That is, adequate availability may be still assumed such that these systems would continue to be available to provide their assumed [safety] function for limiting the dose consequences of an FHA in accordance with the accident analysis currently described in the FSAR [Callaway Final Safety Analysis Report].

With respect to the allowed outage time (Completion Time) for an inoperable EES train, the consequences of a postulated accident are not affected by equipment allowed outage times as long as adequate equipment availability is maintained. The proposed EES allowed outage time is based on the allowed outage time specified in the Standard Technical Specifications (STS) for which it may be presumed that the specified allowed outage time (Completion Time) is acceptable and supports adequate EES availability. As noted in the STS Bases, the 7-day Completion Time for restoring an inoperable EES train takes into account the availability of the other train [(*i.e.*, the other train is operable)]. Since the STS-supported Completion Time supports adequate EES availability, it may be assumed that the EES function would be available for mitigation of an FHA, thus limiting offsite dose to within the currently calculated [dose consequence] values based on the current accident analysis [in the FSAR]. On this basis, the consequences of applicable, [previously] analyzed accidents (*i.e.*, the FHA) are not increased by the proposed change.

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

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

Response: No.

The proposed changes do not create any new failure modes for any system or component, nor do they adversely affect plant operation. No hardware or design changes are involved. Thus, no new equipment will be added and no new limiting single failures must be postulated. The plant will continue to be operated within the envelope of the existing safety analysis [in the FSAR].

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

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

Response: No.

The calculated radiological dose consequences per the applicable accident analyses remain bounding since they are not impacted by the proposed changes. The margins [of safety] to the limits of 10 CFR 100 [Title 10 of the Code of Federal Regulations Part 100] and GDC [General Design Criterion] 19 [of Appendix A to 10 CFR Part 50] are thus unchanged by the proposed changes.

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

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

Attorney for licensee: John O'Neill, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, NW., Washington, DC 20037.

NRC Branch Chief: David Terao.

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

Date of amendment request: May 22, 2006.

Description of amendment request: The proposed amendment revised Technical Specification (TS) 1.1, ''Definitions,'' TS 3.4.13, ''RCS Operational LEAKAGE,'' TS 5.5.8, ''Steam Generator (SG) Program,'' and TS 5.6.7, ''Steam Generator Tube Inspection Report,'' and adds TS 3.4.20, ''Steam Generator (SG) Tube Integrity.'' The proposed changes are necessary in order to implement the guidance for the industry initiative on Nuclear Energy Institute (NEI) 97–06, ''Steam Generator Program Guidelines.'' The licensee has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92, ''Issuance of Amendment,'' as discussed below:

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.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

NRC Branch Chief: Evangelos C. Marinos.

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.*

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

Date of application for amendment: February 6, 2006, as supplemented by letter dated May 5, 2006.

Brief description of amendment: The proposed amendment added a license condition to extend certain Technical Specification (TS) surveillance intervals on a one-time basis to account for the effects of an extended forced outage in the spring of 2005.

Date of issuance: July 12, 2006.

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

Amendment No.: 187.

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

Date of initial notice in the **Federal Register:** March 14, 2006 (71 FR 13172).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 12, 2006.

No significant hazards consideration comments received: No.

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

Date of application of amendments: June 15, 2005.

Brief description of amendments: The amendments revised the Technical Specifications to eliminate the out of date requirements associated with the completion of the Keowee Refurbishment modifications on both Keowee Hydro Units (KHUs).

Date of Issuance: July 11, 2006.

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

Amendment Nos.: 353, 355, and 354.

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

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

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 5, 2005, as supplemented by letter dated March 22, 2006.

Brief description of amendment: The amendment modified the existing Technical Specification 3.3.1.3, ''Oscillation Power Range Monitor (OPRM) Instrumentation,'' Surveillance Requirement 3.3.1.3.5. Specifically, the thermal power level at which the OPRMs are ''not bypassed'' (enabled to perform their design function) will be change from > 28.6-percent rated thermal power to ≥ 23.8-percent rated thermal power.

Date of issuance: June 30, 2006.

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

Amendment No.: 138.

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

Date of initial notice in the **Federal Register:** August 16, 2005 (70 FR 48206).

The March 22, 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 amendment is contained in a Safety Evaluation dated June 30, 2006.

No significant hazards consideration comments received: No.

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 application for amendments: March 7, 2006.

Brief description of amendments: The amendments revised Section 5.5.2, ''Leakage Monitoring Program,'' of the units'' Technical Specifications, adding the Liquid Waste Disposal System, Waste Gas System, and Post-Accident Containment Hydrogen Monitoring System to the list of systems. The listing of these systems was inadvertently omitted from Section 5.5.2.

Date of issuance: July 5, 2006.

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

Amendment Nos.: 294 and 297.

Facility Operating License Nos. DPR–58 and DPR–74: Amendments revise the Technical Specifications and Licenses.

Date of initial notice in the **Federal Register:** April 11, 2006 (71 FR 18374).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 29, 2005, as supplemented by letter dated April 25, 2006.

Brief description of amendment: The amendment revised Technical Specifications Table 3.3.8.1–1, ''Loss of Power Instrumentation,'' changing the allowable values for the 4.16-kV essential bus degraded voltage from a range of 3897–3933 volts to a range of 3913–3927 volts.

Date of issuance: July 3, 2006.

Effective date: As of the date of issuance and shall be implemented concurrently with implementation of the Improved Technical Specifications (Amendment No. 146, dated June 5, 2006).

Amendment No: 147.

Facility Operating License No. DPR–22: Amendment revised the Facility Operating License and Technical Specifications.

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

Date of initial notice in the **Federal Register:** November 23, 2005 (70 FR 70889).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 3, 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: February 16, 2006.

Brief description of amendment: The amendment revised the Technical Specifications to make the existing SG tube surveillance program consistent with the Commission's approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–449, ''Steam Generator Tube Integrity,'' Revision 4.

Date of issuance: July 6, 2006.

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

Amendment No.: 223.

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

Date of initial notice in the **Federal Register:** May 23, 2006 (71 FR 29679).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 6, 2006.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 11, 2005, supplemented by letter dated March 23, 2006.

Brief description of amendments: The amendments revise PINGP's Technical Specification (TS) 3.6.5, ''Containment Spray and Cooling Systems,'' to incorporate changes to an existing condition and two surveillance requirements, and also to add a new condition that will allow continued plant operation with TS limitations when two containment cooling system fan coil units, one in each train, are inoperable.

Date of issuance: June 29, 2006.

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

Amendment Nos.: 173 and 163.

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

Date of initial notice in the **Federal Register:** February 28, 2006 (71 FR 10074).

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

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 29, 2006.

No significant hazards consideration comments received: No.

Pacific Gas and Electric Company, Docket No. 50–133, Humboldt Bay Power Plant, Unit 3, Humboldt County, California

Date of application for amendment: January 19, 2006.

Brief description of amendment: The amendment revises the Humboldt Bay Unit 3 Technical Specifications to correct an editorial error and to allow leaving the Unit 3 control room temporarily unmanned during emergency conditions requiring personnel to evacuate occupied buildings for their safety.

Date of issuance: July 10, 2006.

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

Amendment No.: 38.

Facility Operating License No. DPR–7: This amendment revised the Technical Specifications and License.

Date of initial notice in the **Federal Register:** February 28, 2006 (71 FR 10077).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 10, 2006.

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: February 1, 2006, as supplemented on June 27, 2006.

Brief description of amendments: The amendments revise the Technical Specification (TS) requirements for inoperable snubbers by adding limiting condition for operation 3.0.8 for SSES 1 and 2. This change is based on the TS Task Force (TSTF) change 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 November 24, 2004, and May 4, 2005.

Date of issuance: July 7, 2006.

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

Amendment Nos.: 236 and 213.

Facility Operating License Nos. NPF–14 and NPF–22: The amendments revised the Technical Specifications and License.

Date of initial notice in the **Federal Register:** April 25, 2006 (71 FR 23959).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 7, 2006.

The supplement dated June 27, 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.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendments request: October 6, 2005, as supplemented April 17, 2006.

Brief Description of amendments: The amendments revised Technical Specification (TS) Section 5.6.5, ''Core Operating Limits Report (COLR),'' to reflect the addition of the methodology in WCAP–16009–P–A, ''Realistic Large Break LOCA [Loss-Of-Coolant Accident] Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM),'' for and provide a new large break LOCA analyses for Farley Units 1 and 2.

Date of issuance: July 11, 2006.

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

Amendment Nos.: 174/167.

Renewed Facility Operating License Nos. NPF–2 and NPF–8: Amendments revise the Technical Specifications and Licenses.

Date of initial notice in the **Federal Register:** November 8, 2005 (70 FR 67751). The supplemental letter provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Unit Nos. 1 and 2, Houston County, Alabama

Date of amendments request: February 17, 2006.

Brief Description of amendments: The amendments revised the Technical Specifications (TSs) adding Limiting Condition for Operation (LCO) 3.0.8 to allow 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 consistent with the program in place for complying with the requirements of 10 CFR 50.65(a)(4).

Date of issuance: June 29, 2006.

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

Amendment Nos.: 173/166.

Renewed Facility Operating License Nos. NPF–2 and NPF–8: Amendments revised the Licenses and the Technical Specifications.

Date of initial notice in the **Federal Register:** April 25, 2006 (71 FR 23960).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 29, 2006.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: December 16, 2005.

Brief description of amendments: The amendments revised the Technical Specifications ACTIONS NOTE for TS 3.7.5, ''Auxiliary Feedwater (AFW) System,'' based on Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF–359, Revision 9, ''Increased Flexibility in Mode Restraints.''

Date of issuance: July 14, 2006.

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

Amendment Nos.: 142 and 122.

Facility Operating License Nos. NPF 68 and NPF–81: Amendments revised the Licenses and the Technical Specifications.

Date of initial notice in the **Federal Register:** February 14, 2006 (71 FR 7813).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 14, 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: May 26, 2005, as supplemented by letter dated March 9, 2006.

Brief description of amendment: The amendment revised TS 3.7.2, ''Main Steam Isolation Valves (MSIVs),'' by adding the MSIV actuator trains to (1) the limiting condition for operation (LCO) and (2) the conditions, required actions, and completion times for the LCO. The existing conditions and required actions in TS 3.7.2 are renumbered to account for the new conditions and required actions.

Date of issuance: June 16, 2006.

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

Amendment No.: 172.

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

Date of initial notice in the **Federal Register:** June 21, 2005 (70 FR 35740).

The supplemental letter dated March 9, 2006, provided additional clarifying information, 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 published in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 16, 2006.

No significant hazards consideration comments received: No.

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

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

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

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

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

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

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

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

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

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

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

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

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

1. Technical—primarily concerns/issues relating to technical and/or health and safety matters discussed or referenced in the applications.

2. Environmental—primarily concerns/issues relating to matters discussed or referenced in the environmental analysis for the applications.

3. Miscellaneous—does not fall into one of the categories outlined above.

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

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

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

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

Exelon Generation Company, LLC, Docket No. 50–353, Limerick Generating Station (LGS), Unit 2, Montgomery County, Pennsylvania

Date of amendment request: June 9, 2006, as supplemented June 16, and June 23, 2006.

Description of amendment request: The one-time amendment revises Technical Specification (TS) Limiting Condition for Operation 3.6.1.7 concerning drywell average air temperature. Specifically, the proposed change would add a footnote to the TS limit for drywell average air temperature of 145 degrees Fahrenheit (°F) to allow continued operation of LGS, Unit 2, with drywell average air temperature no greater than 148 °F for the remainder of the current operating cycle (Cycle 9), which is currently scheduled to end in March 2007, or until the next shutdown of sufficient duration to allow for unit cooler fan repairs, whichever comes first.

Date of issuance: July 7, 2006.

Effective date: As of date of issuance, to be implemented within 14 days.

Amendment No.: 145.

Facility Operating License No. NPF–85: The amendment revises the Technical Specifications and License.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. June 20, 2006 (71 FR 35453). 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 by July 5, 2006, 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 July 7, 2006.

The supplements dated June 16 and June 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.

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

NRC Branch Chief: Darrell J. Roberts.

Dated at Rockville, Maryland, this 25th day of July, 2006.

[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.

For the Nuclear Regulatory Commission.

Cornelius F. Holden,

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

[FR Doc. 06–6597 Filed 7–31–06; 8:45 am]

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OFFICE OF THE UNITED STATES TRADE REPRESENTATIVE

Generalized System of Preferences (GSP): Notice of Difficulty in Receiving Petitions for the 2006 Annual GSP Product and Country Practices Review

AGENCY: Office of the United States Trade Representative.

ACTION: Notice of difficulty in receiving petitions for the 2006 Annual GSP Product and Country Practices Review.

SUMMARY: This notice identifies those petitions that the Office of the United States Trade Representative (USTR) received by the deadline of July 20, 2006, for consideration in the 2006 Annual Review. Because of technical difficulties in receiving petitions, USTR requests parties who submitted petitions prior to July 20, 2006, to review the list of petitioners included in the **SUPPLEMENTARY INFORMATION** and to notify the USTR of any petitions that were submitted to the GSP Subcommittee by 5 p.m., July 20, 2006, but not included in that list.

FOR FURTHER INFORMATION CONTACT: The GSP Subcommittee of the Trade Policy Staff Committee, Office of the United States Trade Representative, 1724 F Street, NW., Room F–220, Washington, DC 20508. The telephone number is (202) 395–6971, the facsimile number is (202) 395–9481, and the e-mail address is *FR0618@USTR.EOP.GOV.*

SUPPLEMENTARY INFORMATION: On June 29, 2006, USTR published a request for petitions for the 2006 Annual GSP Product and Country Practices Review (71 FR 37129, June 29, 2006). Because of technical problems, USTR may not have received all the petitions which were submitted. We did receive petitions from the following parties: ANFACER (Brazilian Association of Ceramic Tile Manufacturers), The Home Depot, the International Intellectual Property Association (IIPA), AFL–CIO, and R&J Trading International Company, Inc. Parties that can verify submission of a petition not included in this list should call the GSP Subcommittee at (202) 395–6971 and then resubmit the petition to *FR0618@USTR.EOP.GOV.* Parties must also include proof that the petition was transmitted by e-mail to the GSP Subcommittee by the July 20, 2006, deadline. Such documentation may include a copy of the original e-mail transmitting the petition, indicating the original date and time, from a ''sent message'' folder. The deadline for re-submitting any petitions meeting these criteria is 5 p.m., August 11, 2006.

Public Review: Public versions of all documents relating to the 2006 Annual Review will be available for examination on or before August 21, 2006, by appointment, in the USTR public reading room, 1724 F Street, NW., Washington, DC. Appointments may be made from 9:30 a.m. to noon and 1 p.m. to 4 p.m., Monday through Friday, by calling (202) 395–6186.

Marideth Sandler,

Executive Director GSP, Chairman, GSP Subcommittee of the Trade Policy Staff Committee.

[FR Doc. E6–12313 Filed 7–31–06; 8:45 am]

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OFFICE OF THE UNITED STATES TRADE REPRESENTATIVE

Generalized System of Preferences (GSP): Initiation of a Review To Consider the Designation of East Timor as a Least Developed Beneficiary Developing Country Under the GSP

AGENCY: Office of the United States Trade Representative.

ACTION: Notice and solicitation of public comment.

SUMMARY: This notice announces the initiation of a review to consider the designation of East Timor as a least developed beneficiary developing country under the GSP program and solicits public comment relating to the designation criteria. Comments are due on August 25, 2006, in accordance with the requirements for submissions, explained below.

ADDRESSES: Submit comments by electronic mail (e-mail) to: *FR0618@ustr.eop.gov.* For assistance or if unable to submit comments by e-mail, contact the GSP Subcommittee, Office of the United States Trade Representative; USTR Annex, Room F–220; 1724 F Street, NW., Washington, DC 20508 (Tel. 202–395–6971).

FOR FURTHER INFORMATION CONTACT: Contact the GSP Subcommittee, Office of the United States Trade Representative; USTR Annex, Room F–220; 1724 F Street, NW., Washington, DC 20508 (Telephone: 202–395–6971, Facsimile: 202–395–9481).

SUPPLEMENTARY INFORMATION: The GSP Subcommittee of the Trade Policy Staff Committee (TPSC) has initiated a review in order to make a recommendation to the President as to whether East Timor meets the eligibility criteria of the GSP statute, as set out below. After considering the eligibility criteria, the President is authorized to designate East Timor as a least developed beneficiary developing country for purposes of the GSP.

Interested parties are invited to submit comments regarding the eligibility of East Timor for designation as a least developed beneficiary developing country. Documents should be submitted in accordance with the instructions below to be considered in this review.

Eligibility Criteria

The trade benefits of the GSP program are available to any country that the President designates as a GSP ''beneficiary developing country.'' Additional trade benefits under the GSP are available to any country that the President designates as a GSP ''least-developed beneficiary developing country.'' In designating countries as GSP beneficiary developing countries, the President must consider the criteria in sections 502(b)(2) and 502(c) of the Trade Act of 1974, as amended (19 U.S.C. 2462(b)(2), 2462(c)) (''the Act''). Section 502(b)(2) provides that a country is ineligible for designation if:

1. Such country is a Communist country, unless—

(a) The products of such country receive nondiscriminatory treatment, (b) Such country is a WTO Member (as such term is defined in section 2(10) of the Uruguay Round Agreements Act) (19 U.S.C. 3501(10)) and a member of the International Monetary Fund, and (c) Such country is not dominated or controlled by international communism.

2. Such country is a party to an arrangement of countries and participates in any action pursuant to such arrangement, the effect of which is—

(a) To withhold supplies of vital commodity resources from international trade or to raise the price of such commodities to an unreasonable level, and (b) To cause serious disruption of the world economy.

3. Such country affords preferential treatment to the products of a developed country, other than the United States, which has, or is likely to have, a significant adverse effect on United States commerce.

4. Such country—

(a) Has nationalized, expropriated, or otherwise seized ownership or control of property, including patents, trademarks, or copyrights, owned by a