prior to the meeting to be advised of any potential changes in the agenda.

Dated: May 6, 2003.

**Sher Bahadur,**

*Associate Director for Technical Support, ACRS/ACNW.*

[FR Doc. 03–11838 Filed 5–12–03; 8:45 am]

**BILLING CODE 7590–01–P**

## NUCLEAR REGULATORY COMMISSION

### Sunshine Act, Meeting

**DATE:** Weeks of May 12, 19, 26, June 2, 9, 16, 2003.

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

**STATUS:** Public and Closed.

**MATTERS TO BE CONSIDERED:**

*Week of May 12, 2003*

Wednesday, May 14, 2003

1:30 p.m. Discussion of Security Issues (Closed—Ex. 1)

Thursday, May 15, 2003

9:30 a.m. Briefing on Results of Agency Action Review Meeting (Public Meeting) (Contact: Robert Pascarelli, 301–415–1245). Morning session.

12:30 p.m. Briefing on Results of Agency Action Review Meeting (Public Meeting) (Contact: Robert Pascarelli, 301–415–1245). Afternoon session.

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

*Week of May 19, 2003—Tentative*

There are no meetings scheduled for the Week of May 19, 2003.

*Week of May 26, 2003—Tentative*

Wednesday, May 28, 2003

9:30 a.m. Meeting with Advisory Committee on the Medical Uses of Isotopes (ACMUI) (Public Meeting) (Contact: Angela Williamson, 301–415–5030)

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

2:45 p.m. Discussion of Management Issues (Closed—Ex. 2)

Thursday, May 29, 2003

9:30 a.m. Briefing on Status of Revisions to the Regulatory Framework for Steam Generator Tube Integrity (Public Meeting) (Contact: Louise Lund, 301–415–3248)

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

2 p.m. Briefing on Equal Employment Opportunity Program (Public Meeting) (Contact: Corenthis Kelley, 301–415–7380)

*Week of June 2, 2003—Tentative*

There are no meetings scheduled for the Week of June 2, 2003.

*Week of June 9, 2003—Tentative*

Wednesday, June 11, 2003

10:30 a.m. All Employees Meeting.
1:30 p.m. All Employees Meeting.

*Week of June 16, 2003—Tentative*

There are no meetings scheduled for the Week of June 16, 2003.

*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: David Louis Gamberoni (301) 415–1651.

* * * * *

**ADDITIONAL INFORMATION:** By a vote of 4–0 on April 28, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Affirmation of Nuclear Fuel Services, Inc. (Erwin, Tennessee)'' be held on April 29, and on less than one week's notice to the public.

* * * * *

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

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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: May 8, 2003.

**D.L. Gamberoni,**

*Technical Coordinator, Office of the Secretary.*

[FR Doc. 03–11962 Filed 5–9–03; 10:07 am]

**BILLING CODE 7590–01–M**

## NUCLEAR REGULATORY COMMISSION

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

#### I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from, April 18, 2003, through May 1, 2003. The last biweekly notice was published on April 29, 2003 (68 FR 22744).

#### 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.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing 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 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By June 12, 2003, 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 2.714, 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 by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, 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 factors: (1) The nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of 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. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies 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, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff, or may be delivered to the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. Because of continuing disruptions in delivery of mail to United States Government offices, it is requested that petitions for leave to intervene and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301–415–1101 or by e-mail to *hearingdocket@nrc.gov.* 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 because of continuing disruptions in delivery of mail to United States Government offices, 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 filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

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 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 NRC PDR Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

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

*Date of amendment request:* April 2, 2001, as supplemented by letters dated January 15, August 23, 2002, and March 28, 2003.

*Description of amendment request:* The proposed amendment would add operational restrictions when the inclined fuel transfer system (IFTS) blind flange is removed during Modes 1, ''Power Operation,'' 2, ''Startup,'' or 3, ''Hot Shutdown.'' The proposed changes would (1) include a limitation on the duration that the IFTS blind flange can be removed while primary containment integrity is required, (2) include a limitation on the duration that the IFTS blind flange can remain in the unbolted configuration, (3) specify the need to install the steam dryer pool to reactor cavity pool gate prior to opening the blind flange, and (4) provide the flexibility to remove the IFTS blind flange for other than maintenance and testing purposes only.

*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?

The proposed changes allow operation of the IFTS while primary containment operability is required. The proposed changes result in a change to the primary containment boundary. A loss of primary containment integrity is not an accident initiator. The proposed changes do not involve any modifications to plant systems or design parameters or conditions that contribute to the initiation of any accidents previously evaluated. Therefore, the proposed changes do not increase the probability of any accident previously evaluated.

The proposed changes potentially affect the allowable leakage of the containment structure which is designed to mitigate the consequences of a loss-of-coolant accident (LOCA). The function of the primary containment is to maintain functional integrity during and following the peak transient pressures and temperatures that result from any LOCA. The primary containment is designed to limit fission product leakage following the design basis LOCA. Because the proposed changes do not alter the plant design, only the extent of the boundaries that provide primary containment isolation for the IFTS penetration, the proposed changes do not result in an increase in primary containment leakage. In addition, a time limit for IFTS blind flange removal of 40 days per cycle and a 12 hour limit for the unbolted configuration of the IFTS flange have been established as conservative measures to limit the associated risk to the containment boundary for all accident conditions. Once the blind flange is removed the IFTS transfer tube and its appurtenances become part of the primary containment boundary. As part of the primary containment boundary these subject components would be exposed to LOCA pressures. While these components have not been fabricated or installed to meet the acceptance criteria for a containment penetration, they have been built to withstand the rigors of a commercial nuclear application. This includes, but is not limited to, consideration of adequate seismic support, inertial forces imparted to the fuel, appropriate cooling and shielding for the spent nuclear fuel, integrity of the fluid system pressure boundary, and a safety analysis, including a failure modes and effects evaluation which assumes that credible events and credible combinations of events have been considered and mitigated against by either a fail safe design or redundancy. They are judged to be an acceptable barrier to prevent the uncontrolled release of post-accident fission products for the purposes of this amendment request.

Further, it has been shown that the largest potential leakage pathway, the IFTS transfer tube itself, would remain sealed by the depth of water required by the proposed [technical specification] TS change to be maintained in the fuel building fuel transfer pool. The transfer tube drain line constitutes the other possible leakage pathway, and will be required to be capable of being isolated via administrative control of the manual isolation valve in the drain line. Additionally, due to the physical relationships of the buildings and components involved, any leakage from either of these pathways is fully contained within the boundaries of the secondary containment and would be filtered by the Standby Gas Treatment System prior to release to the environment.

Leakage from the containment upper pool through the open IFTS transfer tube could potentially result in the excessive loss of water from the volume intended to provide post-LOCA makeup water to the suppression pool. The upper pool dump volume is maintained by requiring the installation of the steam dryer pool to reactor cavity pool gate with the seal inflated and a backup air supply provided. Maintaining the upper pool dump volume ensures proper suppression pool level can be achieved following a LOCA which provides for long-term steam condensation.

Based on the above, the proposed changes do not increase the consequences of an accident previously evaluated.

In summary, 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 involve a change to the plant design or operation except for when IFTS is operated. As a result, the proposed changes do not affect any of the parameters or conditions that could contribute to the initiation of any accidents. No new accident modes or equipment failure modes are created by these changes. Extending the primary containment boundary to include portions of the IFTS has no influence on, nor does it contribute to the possibility of a new or different kind of accident or malfunction from those previously evaluated. Furthermore, operation of IFTS is unrelated to the operation of the reactor. There is no mishap in the process that can lead or contribute to the possibility of losing any coolant in the reactor or introducing the chance for positive or negative reactivity or other accidents different from and not bounded by those previously evaluated. Therefore, these 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 only affect the extent of a portion of the primary containment boundary. The time that the IFTS is in the seismically indeterminate configuration with the flange unbolted will be limited to 12 hours per operating cycle. The time the IFTS blind flange will be removed will be limited to 40 days per operating cycle. These restrictions will limit the risk from the potential leakage through the primary containment boundary. Having IFTS in operation does not affect the reliability of equipment used for core cooling. In addition, precautions will be taken to administratively control the IFTS transfer tube drain path so that the proposed change will not increase the probability that an increase in leakage from the primary containment to the secondary containment could occur. Precautions will also be taken to ensure that the steam dryer pool to reactor cavity pool gate is installed prior to removing the IFTS flange when primary containment is required to be operable. Installation of this gate will ensure that an adequate containment upper pool dump volume is maintained to support post-LOCA suppression pool makeup water volume requirements.

The margin of safety that has the potential of being impacted by the proposed changes involve the offsite dose consequences of postulated accidents which are directly related to containment leakage rate. The containment isolation system is designed to limit leakage to $L_a$ which is defined by the

[Clinton Power Station] CPS TS to be 0.65% of primary containment air weight per day at the design basis LOCA maximum peak containment pressure (*i.e.*, $P_a$). The limitation on containment leakage rate is designed to ensure that total leakage volume will not exceed the volume assumed in the accident analyses at $P_a$. The margin of safety for the offsite dose consequences of postulated accidents directly related to the containment leakage rate is maintained by meeting the $L_a$ acceptance criteria during operation. The $L_a$ value is not being modified by this proposed TS change. The IFTS will continue to provide an acceptable barrier to prevent unacceptable containment leakage during a LOCA, and therefore these changes will not create a situation causing the containment leakage rate acceptance criteria to be violated.

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:* Edward J. Cullen, Deputy General Counsel Exelon BSC—Legal, 2301 Market Street, Philadelphia, PA 19101.

*NRC Section Chief:* Anthony J. Mendiola.

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

*Date of amendments request:* March 28, 2003.

*Description of amendments request:* The amendment would remove the post-accident hydrogen monitoring and control requirements from the Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 Technical Specifications.

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

The proposed change to the Technical Specifications has been evaluated against the standards in 10 CFR 50.92. The proposed amendment revises Technical Specification 3.3.10, Post-Accident Monitoring Instrumentation, and Technical Specification Table 3.3.10–1, Post-Accident Monitoring Instrumentation to delete references to the containment hydrogen analyzers. Additionally, the proposed amendment will delete Technical Specification 3.6.7, Hydrogen Recombiners. The proposed change has been determined to not involve a significant hazards consideration, in that operation of the facility in accordance with the proposed amendments:

1. Would not involve a significant increase in the probability or consequences of any accident previously evaluated.

Components used in the control of hydrogen in the Containment (consisting of hydrogen recombiners, a hydrogen vent, and hydrogen detectors) are not considered accident initiators. Therefore, this change does not increase the probability of an accident previously evaluated.

The purpose of the Hydrogen Control System is to ensure that hydrogen concentration is maintained below 4.0 volume percent so that Containment integrity is not challenged following a design basis loss-of-coolant accident (LOCA). The Calvert Cliffs Nuclear Power Plant Individual Plant Examination analyzed the probability of Containment failure under a variety of conditions. This proposed amendment does not alter the conclusions or assumptions of the Individual Plant Examination. The Calvert Cliffs Nuclear Power Plant Containment provides a safety margin against hydrogen burn following a design basis accident, such that the Containment will not fail even without hydrogen control equipment. Therefore, this change does not increase the consequences of accidents previously evaluated.

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

2. Would not create the possibility of a new or different [kind] of accident from any accident previously evaluated.

The proposed change does not change the configuration of the plant beyond the Hydrogen Control System. Hydrogen generation following a design basis LOCA has been evaluated. Deletion of the Hydrogen Control System from the plant design basis and Technical Specifications does not alter the generation of hydrogen post-LOCA.

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

3. Would not involve a significant reduction in [a] margin of safety.

The margin of safety in this case is the ability of Containment to withstand a pressure increase caused by the deflagration of hydrogen in the Containment. Industry experience and experimentation has shown that large, dry, well-ventilated Containments such as those at Calvert Cliffs can withstand pressures generated by ignition of hydrogen resulting from a LOCA. The Calvert Cliffs Nuclear Power Plant Containment provides a safety margin against hydrogen burn following a design basis accident, such that the Containment will not fail even without hydrogen control equipment.

Therefore, this change does not significantly reduce [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 amendments request involves no significant hazards consideration.

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

*NRC Section Chief:* Richard J. Laufer.

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

*Date of amendment request:* April 2, 2003.

*Description of amendment request:* The proposed amendment would change the spent fuel pool loading restrictions by redefining the regions, inserting Metamic® poison panels in a portion of the spent fuel pool, and increasing the minimum boron concentration.

*Basis for 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 three fuel handling accidents described below can be postulated to increase reactivity. However, for these accident conditions, the double contingency principle of ANS [American Nuclear Society] N16.1–1975 is applied. This states that it is unnecessary to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for accident conditions, the presence of soluble boron in the storage pool water can be assumed as a realistic initial condition since its absence would be a second unlikely event.

Three types of drop accidents have been considered: a vertical drop accident, a horizontal drop accident, and an inadvertent drop of an assembly between the outside periphery of the rack and the pool wall:

- A vertical drop directly upon a cell will cause damage to the racks in the active fuel region. The current 1600 ppm soluble boron concentration TS limit will ensure that $K_{eff}$ does not exceed 0.95.
- A fuel assembly dropped on top of the rack horizontally will not deform the rack structure such that criticality assumptions are invalidated. The rack structure is such that an assembly positioned horizontally on top of the rack results in a separation distance from the upper end of the active fuel region of the stored assemblies. This distance is sufficient to preclude interaction between the dropped assembly and the stored fuel.
- An inadvertent drop of an assembly between the outside periphery of the rack and the pool wall is bounded by the worst case fuel misplacement accident condition.

The fuel assembly misplacement accident was considered for all storage configurations. An assembly with high reactivity is assumed to be placed in a storage location which requires restricted storage based on initial U–235 loading, cooling time, and burnup. The presence of boron in the pool water assumed in the analysis has been shown to offset the worst case reactivity effect of a misplaced fuel assembly for any configuration. This boron requirement is less than the 1600 ppm currently required by the ANO–1 TS. Thus, a five percent subcriticality margin can be easily met for postulated accidents, since any reactivity increase will be much less than the negative worth of the dissolved boron.

For fuel storage applications, water is usually present. An ''optimum moderation'' accident is not a concern in spent fuel pool storage racks because the rack design prevents the preferential reduction of water density between the cells of a rack (*e.g.*, boiling between cells). An ''optimum moderation'' accident in the new fuel pit was previously evaluated and the conclusions of that evaluation have not changed as a result of the fuel enrichment.

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

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

*Response:* No.

The proposed changes will define a portion of the current Region 2 as Region 3. The new region will contain Metamic® poison panel inserts and will allow unrestricted storage of fuel assemblies with various enrichments and burnup. To support the proposed change, new criticality analyses have been performed. The analyses resulted in new loading restrictions in Region 1 and Region 2. The presence of boron in the pool water assumed in the analysis is less than the 1600 ppm currently required by the ANO–1 TSs.

Thus, a five percent subcriticality margin can be easily met for postulated accidents, since any reactivity increase will be much less than the negative worth of the dissolved boron.

No new or different types of fuel assembly drop scenarios are created by the proposed change. During the installation of the Metamic® panels, the possible drop of a panel is bounded by the current fuel assembly drop analysis. No new or different fuel assembly misplacement accidents will be created. Administrative controls currently exist to assist in assuring fuel misplacement does not occur.

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.

With the presence of a nominal boron concentration, the SFP storage racks will be designed to assure that fuel assemblies of less than or equal to five weight percent U–235 enrichment when loaded in accordance with the proposed loading restrictions will be maintained within a subcritical array with a five percent subcritical margin (95% probability at the 95% confidence level). This has been verified by criticality analyses.

Credit for soluble boron in the SFP water is permitted under accident conditions. The proposed modification that will allow insertion of Metamic® poison panels does not result in the potential of any new misplacement scenarios. Criticality analyses have been performed to determine the required boron concentration that would ensure the maximum $K_{eff}$ does not exceed 0.95. The ANO–1 TS for the minimum SFP boron concentration is greater than that required to ensure $K_{eff}$ does not exceed 0.95. Therefore, the margin of safety currently defined by taking credit for soluble boron will be maintained.

The structural analysis of the spent fuel racks, along with the evaluation of the SFP structure, showed that the integrity of these structures will be maintained with the addition of the poison inserts. The structural requirements were shown to be satisfied, so the safety margins were maintained.

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

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

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

*NRC Section Chief:* Robert A. Gramm.

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

*Date of amendment request:* March 19, 2003.

*Description of amendment request:* The proposed amendment deletes requirements from the technical specifications (TS) and other elements of the licensing bases to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI Unit 2 (TMI–2). Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The changes are based on U.S. Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–413, ''Elimination of Requirements for a Post Accident Sampling System (PASS).'' The NRC staff issued a notice of opportunity for comment in the **Federal Register** on December 27, 2001 (66 FR 66949), on possible amendments concerning TSTF–413, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 20, 2002 (67 FR 13027). The licensee affirmed the applicability of the following NSHC determination in its application dated March 19, 2003.

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

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

The PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the TMI–2 accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections. The system was not intended to and does not serve a function for preventing accidents and its elimination would not affect the probability of accidents previously evaluated.

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a PASS provides little actual benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of

Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

The regulatory requirements for the PASS can be eliminated without degrading the plant 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. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post–TMI–2 accident guidance through the use of the SAMGs, 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 PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in 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 PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radioisotopes 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 the Margin of Safety.

The elimination of the PASS, 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. Methodologies that are not reliant on PASS are designed 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 use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI–2 accident can be adequately met without reliance on a PASS.

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

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

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

*NRC Section Chief:* Robert A. Gramm.

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

*Date of amendment request:* April 3, 2003.

*Description of amendment request:* The proposed changes will revise the Updated Final Safety Analysis Report to change the Reactor Vessel Material Surveillance Program. The change will reflect participation in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP).

*Basis for proposed no significant hazards consideration determination:* As required by Section 50.91(a) of Title 10 of the Code of Federal Regulations (10 CFR ), 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.

Changes in the fracture toughness properties of reactor vessel beltline materials, resulting from the neutron irradiation and the thermal environment, are monitored by a surveillance program in compliance with the requirements of 10CFR50, Appendix H. The proposed change implements an integrated surveillance program that has been evaluated by the NRC [U.S. Nuclear Regulatory Commission] staff as meeting the requirements of paragraph III.C of Appendix H to 10 CFR 50. The BWRVIP's ISP surveillance material selection process adequately ensures that materials in the program effectively provide meaningful information to monitor changes in fracture toughness for GGNS [Grand Gulf Nuclear Station, Unit 1, or Grand Gulf] RPV [Reactor Pressure Vessel] materials. In addition, the ISP program requires participants to acquire and evaluate relevant ISP test data from the program which may affect RPV integrity evaluations in a timely manner. One advantage of participating in the BWRVIP ISP is that surveillance test data applicable to the Grand Gulf RPV will be available sooner than under the current plant specific program.

The proposed change will not affect current RPV performance and will not cause the RPV or interfacing systems to be operated outside of their design or testing limits. The proposed change will not alter any assumptions previously made in evaluating the radiological consequences of accidents.

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

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

*Response:* No.

The change does not affect the design, function, reliability, or operation of any plant structure, system or component. The purpose of the reactor vessel material surveillance program is to monitor neutron embrittlement and thermal environment effects in order to predict the behavioral characteristics of materials of pressure retaining components of the reactor coolant pressure boundary and to ensure that reactor vessel fracture toughness and integrity requirements are not violated. The ISP is an approved alternate monitoring program that meets the regulatory requirements in Appendix H to 10 CFR 50. As an acceptable alternate monitoring program, the ISP cannot create a new failure mode involving the possibility of a new or different kind of accident.

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

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

*Response:* No.

The reactor material surveillance program required by 10 CFR 50, Appendix H, is designed to ensure that adequate margins of safety are provided for the reactor coolant pressure boundary during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. Monitoring changes in the fracture toughness of reactor vessel materials ensures that material changes due to radiation embrittlement are adequately considered for safe reactor operations. Paragraph lll.C of Appendix H to 10 CFR 50 delineates the regulatory requirements for an ISP. The BWRVIP ISP meets these requirements and has been approved by the NRC.

One of the uses of the material surveillance data obtained through the proposed ISP is to ensure the reactor coolant system P/T [Pressure/Temperature] limits established by the Technical Specifications are conservative. The material surveillance data obtained through the proposed Integrated Surveillance Program will provide new information that will be evaluated to ensure that the P/T limits are conservative. In addition, a neutron fluence calculation methodology which has been approved by the NRC staff and is consistent with the attributes identified in U.S. Nuclear Regulatory Commission Regulatory Guide 1.190, ''Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,'' will be used for the determination of reactor vessel and surveillance capsule neutron fluence values to ensure quality of the method and compatibility between ISP results.

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

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

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

*NRC Section Chief:* Robert A. Gramm.

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

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

*Date of application for amendment request:* February 14, 2003.

*Description of amendment request:* The proposed amendments would relax the Technical Specifications (TSs) surveillance requirement (SR) for reactor instrumentation line excess flow check valves (EFCVs). Currently, TSs require testing of each reactor instrumentation line EFCV on a 24-month frequency. The proposed TS SR would require that a representative sample of reactor instrumentation line EFCVs be tested every 24 months, such that each EFCV will be tested nominally once every 10 years.

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

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

The current Technical Specification (TS) Surveillance Requirement (SR) frequency requires each reactor instrumentation line excess flow check valve (EFCV) to be tested every 24 months. The EFCVs at Dresden Nuclear Power Station (DNPS) and Quad Cities Nuclear Power Station (QCNPS) are designed to remain open during normal operation, but will close automatically in the event of an instrument line break downstream of the valve. The proposed change allows a reduced number of reactor instrumentation line EFCVs to be tested every 24 months. Industry operating experience demonstrates a high level of reliability for these EFCVs. A failure of an EFCV to isolate cannot initiate previously evaluated accidents (*i.e.*, a break in a reactor coolant pressure boundary (RCPB) instrument line outside containment). Therefore, there is no increase in the probability of an accident as a result of this proposed change.

The postulated break of an instrument line connected to the RCPB is discussed and evaluated in the Updated Final Safety Analysis Reports (UFSARs) for DNPS and QCNPS. The integrity and functional performance of the secondary containment and standby gas treatment system are not impaired by this event, and the calculated potential offsite exposures are below the guidelines of 10 CFR 100, ''Reactor Site Criteria.'' The NRC approved General Electric Nuclear Energy Licensing Topical Report, NEDO–32977–A, ''Excess Flow Check Valve Testing Relaxation,'' discusses through operating experience that there is a high degree of reliability with the EFCVs and that there are little radiological consequences resulting from an EFCV failure. The radiological consequences for an instrument line break do not credit the EFCVs for isolating the break. Therefore, the consequences of an instrument line break are not impacted by the proposed level of testing. Based on the above, the proposed TS change does not involve a significant increase in the consequences of an accident previously evaluated.

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

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

The proposed change allows a reduced number of reactor instrumentation line EFCVs to be tested every 24 months. No other changes in requirements are being proposed. Industry operating experience as documented in NEDO–32977–A, provides supporting evidence that the reduced testing will not affect the high reliability of these valves. The potential failure of an EFCV to isolate as a result of the proposed reduction in testing is bounded by the evaluation of an instrument line break described in the UFSARs for DNPS and QCNPS. The proposed changes do not physically alter the plant and will not alter the operation of structures, systems, and components as described in the UFSARs. Therefore, a new or different kind of accident from any accident previously evaluated will not be created.

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

The consequences of an unisolable rupture of a RCPB instrument line outside containment has been previously evaluated in the UFSARs for DNPS and QCNPS. That evaluation assumed a continuous discharge of reactor coolant for the duration of the detection and cooldown sequence (*i.e.*, no credit was assumed for isolating the break by the associated EFCV in the ruptured instrument line). Since a continuous discharge was assumed in this evaluation, any potential failure of the associated EFCV to isolate postulated by the reduced testing frequency is bounded. 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. Edward J. Cullen, Vice President, General Counsel, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

*NRC Section Chief:* Anthony J. Mendiola.

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

*Date of amendment request:* March 31, 2003.

*Description of amendment request:* The proposed amendments would revise Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF–11 and NPF–18. Specifically, the proposed change will increase the upper limit associated with TS Table 3.3.5.1–1, ''Emergency Core Cooling System Instrumentation,'' Function 3.e, ''HPCS System Flow Rate—Low (Bypass),'' Allowable Value from less than or equal to (≤) 1704 gallons per minute (gpm) to ≤ 2194 gpm. The proposed change increases the Allowable Value band to account for instrumentation deadband, as-left setting tolerances and setpoint drift, and resolves historical difficulties during calibration. The current Allowable Value was initially provided in the LaSalle County Station TS during conversion to Improved Technical Specifications (ITS) format. This value was based on vendor supplied data and believed at the time to adequately account for these parameters. The upper Allowable Value limit is being increased based on historical performance data for the High Pressure Core Spray (HPCS) system flow switches. The increase in the allowed bypass flow rate does not affect the capability of the HPCS system in performing its intended safety function.

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

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

The proposed change to LaSalle County Station Technical Specifications (TS) Table 3.3.5.1–1, ''Emergency Core Cooling System

Instrumentation,'' Function 3.e, ''HPCS System Flow Rate—Low (Bypass),'' request an increase in the Allowable Value from less than or equal to ≤ 1704 gpm to ≤ 2194 gpm. The operation of High Pressure Core Spray (HPCS) System Flow Rate—Low (Bypass) function is not a precursor to any accident previously evaluated. Thus, the proposed change does not have any effect on the probability of an accident previously evaluated.

The LaSalle County Station Emergency Core Cooling Systems (ECCS) are designed, in conjunction with the primary and secondary containments, to limit the release of radioactive material to the environment following a loss of coolant accident (LOCA). The ECCS uses two independent methods, flooding and spraying, to cool the reactor core following a LOCA. The HPCS is one of the core spray systems. The evaluation of the proposed change concluded that the HPCS will operate as assumed in accidents previously evaluated. Thus, the radiological consequences of any accident previously evaluated are not increased.

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

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

The proposed change does not affect the control parameters governing unit operation and does not introduce any new equipment, modes of system operation or failure mechanisms. Calculations have been performed which evaluated the performance of the HPCS system without the closure of the minimum flow bypass valve. The calculations determined that the Unit 1 and Unit 2 HPCS pump capacity with the minimum flow bypass valve open will support HPCS System injection flow into the reactor pressure vessel (RPV) over the full range of RPV pressures above the requirements for HPCS in the Loss of Coolant Accident (LOCA) analyses.

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

3. The proposed changes do not involve a significant reduction in a margin of safety.

The HPCS System Flow Rate—Low (Bypass) Function is one of the inputs to the logic that controls the opening and closing of the minimum flow bypass valve. The current Allowable Values for this function are greater than or equal to (≥)1380 gpm and ≤1704 gpm. The lower Allowable Value limit (*i.e.,* 1380 gpm) ensures that the minimum flow bypass valve opens when pump flow is too low for adequate cooling of the pump while the pump is operating. This limit is not affected by the proposed change.

The upper Allowable Value limit (*i.e.,* 1704 gpm) ensures that the minimum flow bypass valve automatically closes to allow maximum flow to the RPV spray sparger. The proposed change increases the value to ≤ 2194 gpm. LaSalle County Station has evaluated the effect of this change and concluded the following:

- The proposed change to increase the upper Allowable Value limit from ≤ 1704 gpm to ≤ 2194 gpm will provide further assurance that the minimum flow bypass valve remains full open until the HPCS pump flow to the RPV spray sparger is sufficient to prevent overheating of the pump, and
- The upper Allowable Value ensures that the HPCS minimum flow bypass valve closes to allow maximum flow to the RPV spray sparger. The proposed change will delay the initiation of valve closure from ≤ 1704 gpm to ≤ 2194 gpm. The calculations determined that the Unit 1 and Unit 2 HPCS pump capacity with the minimum flow bypass valve open will support HPCS system injection flow into the RPV over the full range of RPV pressures above the requirements for HPCS in the Loss of Coolant Accident (LOCA) analysis up to the maximum assumed injection flow of 5400 gpm. The margin to the flow requirements of the LOCA analysis varies from approximately 200 gpm at very low RPV pressures to greater than 1000 gpm at higher RPV pressures. Since the HPCS system injection flow requirement to the RPV spray sparger assumed in the LOCA analysis is met with the minimum flow bypass valve open, the LOCA analysis results are not adversely affected by increasing the value of flow when the minimum flow bypass valve starts to close. Although the calculations show that closure of the HPCS minimum flow bypass valve is not necessary to meet the HPCS system injection flow requirements assumed in the LOCA analyses, LaSalle County Station has chosen to retain the upper Allowable Value in the TS to provide additional margin to the assumed injection flow of the analyses.

Thus, increasing the TS upper Allowable Value limit for the HPCS System Flow Rate—Low (Bypass) Function from ≤ 1704 gpm to ≤ 2194 gpm will not affect the capability of the HPCS system in performing its intended safety function.

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

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

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

*Attorney for licensee:* Mr. Edward J. Cullen, Deputy General Counsel, Exelon BSC—Legal, 2301 Market Street, Philadelphia, PA 19101.

*NRC Section Chief:* Anthony J. Mendiola.

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

*Date of amendment request:* March 19, 2003.

*Description of amendment request:* The proposed amendment would delete requirements from the Technical Specifications (TSs) and other elements of the licensing bases related to the post-accident sampling system (PASS) at the Monticello Nuclear Generating Plant. Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TSs for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The proposed changes are based on NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–413, ''Elimination of Requirements for a Post Accident Sampling System (PASS).'' The NRC staff issued a notice of opportunity for comment in the **Federal Register** on December 27, 2001 (66 FR 66949), on possible amendments concerning TSTF–413. The notice included a model safety evaluation and model no significant hazards consideration determination, using the consolidated line-item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 20, 2002 (67 FR 13027). The licensee affirmed the applicability of the following no significant hazards consideration determination in its application dated March 19, 2003.

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

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

The PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the TMI–2 accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections. The system was not intended to and does not serve a function for preventing accidents and its elimination would not affect the probability of accidents previously evaluated.

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a PASS provides little actual benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

The regulatory requirements for the PASS can be eliminated without degrading the plant 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. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post-TMI–2 accident guidance through the use of the SAMGs, 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 PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in 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 PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radioisotopes 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 the Margin of Safety

The elimination of the PASS, 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. Methodologies that are not reliant on PASS are designed 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 use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI–2 accident can be adequately met without reliance on a PASS.

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

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

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

*NRC Section Chief:* L. Raghavan.

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

*Date of amendment request:* March 27, 2003.

*Description of amendment request:* The proposed amendment would approve a selective scope application of an alternative source term (AST) for fuel handling accidents (FHAs). Specifically, the amendments would revise Technical Specification (TS) 3.9.3, ''Containment Penetrations,'' to (1) change the Applicability statement to ''During movement of recently irradiated fuel assemblies within containment,'' and (2) modify the Required Action for Condition A to eliminate the requirement to suspend core alterations and add the requirement to suspend movement of recently irradiated fuel assemblies within containment if one or more containment penetrations are not in the required status.

*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. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.

Selective implementation of the Alternative Source Term (AST) and those plant systems affected by implementing the proposed changes to the TS are not accident initiators and cannot increase the probability of an accident. The AST does not adversely affect the design or operation of the facility in a manner that would create an increase in the probability of an accident. Rather, the AST is a methodology used to evaluate the dose consequences of a postulated accident.

The fuel handling accident analysis has demonstrated that the dose consequences of a postulated fuel handling accident remain within the limits provided sufficient decay has occurred prior to the movement of irradiated fuel without taking credit for certain mitigation features such as ventilation filter systems and containment closure. Irradiated fuel that has not undergone the required decay period of 65 hours is defined as recently irradiated fuel and the currently approved TS requirements are applicable when this recently irradiated fuel is being handled.

This amendment does not alter the methodology or equipment used directly in fuel handling operations. Neither ventilation filter system (*i.e.*, the containment purge or drumming area vent stack) is used to actually handle fuel. Neither of these systems is an accident initiator. Similarly, neither the equipment hatch, personnel air locks, any other containment penetrations, nor any component thereof is an accident initiator. No other accident initiator is affected by the proposed changes.

The TEDE [total effective dose equivalent] doses from the analysis supporting this amendment request have been compared to equivalent TEDE doses estimated with the guidelines of RG [Regulatory Guide] 1.183 [''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors''] Footnote 7. The new values are shown to be comparable to the results of the previous analysis.

Based on the aforementioned reasons, the proposed amendment does not involve a significant increase in the probability or consequences of a FHA as previously analyzed.

2. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a new or different kind of accident from any accident previously evaluated.

The evaluation of the effects of the proposed changes indicates that all design standards and applicable safety criteria limits are met. The proposed amendment would increase the time during which the equipment hatch and personnel air locks could be open during core alterations and movement of irradiated fuel. The proposed amendment does not involve changes in the operations of these containment penetrations. Having these penetrations open does not create the possibility of a new accident.

Therefore, operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant reduction in a margin of safety.

The assumptions and input used in the analysis are conservative as noted below. The design basis FHA has been defined to identify conservative conditions. The source term and radioactivity releases have been calculated pursuant to RG 1.183, Appendix B and with conservative assumptions concerning prior reactor operations. The control room atmospheric dispersion factor has been calculated with conservative assumptions associated with the release. The conservative assumptions and input noted above ensure that the radiation doses cited in the amendment request are the upper bound to radiological consequences of a FHA either in containment or in the spent fuel pool. The analysis shows that there is a significant margin between the TEDE radiation doses calculated for the postulated FHA using the AST and acceptance limits of 10 CFR 50.67 and RG 1.183. The proposed changes will not degrade the plant protective boundaries, will not cause a release of fission products to the public, and will not degrade the performance of any Structures, Systems, and Components important to safety. Therefore, there is no significant reduction in the margin of safety as a result of the proposed changes.

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 H. O'Neill, Jr., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

*NRC Section Chief:* L. Raghavan.

*Southern Nuclear Operating Company, Inc, Docket No. 50–364, Joseph M. Farley Nuclear Plant, Unit 2, Houston County, Alabama*

*Date of amendment request:* February 11, 2003.

*Description of amendment request:* The proposed amendments would allow a 40-month inspection interval for Farley, Unit 2 after the completion of the first post-replacement in-service inspection, rather than the completion of two consecutive inspections resulting in a classification of C–1.

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

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

The proposed one-time change revises the steam generator (SG) inspection interval requirements in TS [technical specification] 5.5.9.3, ''Inspection Frequencies,'' for the FNP [Farley Nuclear Plant] Unit 2 Spring 2004 refueling outage, to allow a 40[-]month inspection frequency after one inspection, rather than after two consecutive inspections with results that are within the C–1 category. C–1 category is defined as ''less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.''

The proposed one-time extension of the FNP Unit 2 SG tube inservice inspection interval does not involve changing any structure, system, or component, or affect reactor operations. It is not an initiator of an accident and does not change any existing safety analysis previously analyzed in the FNP's Final Safety Analysis Report (FSAR). As such, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

Since the proposed change does not alter the plant design, there is no direct increase in SG leakage. Industry experience indicates that the probability of increased SG tube degradation would not go undetected. Additionally, steps described below will further minimize the risk associated with this extension. For example, the scope of inspections performed during the last FNP Unit 2 refueling outage (*i.e.*, the first refueling outage following SG replacement) exceeded the TS requirements for the first two refueling outages after SG replacement. That is, more tubes were inspected than were required by TS. Currently, FNP Unit 2 does not have a SG damage mechanism, and will meet the current industry examination guidelines without performing SG inspections during the next refueling outage. Additionally, as part of the FNP SG Program, both a Condition Monitoring Assessment and an Operational Assessment are performed after each inspection and compared to the Nuclear Energy Institute (NEI) 97–06, ''Steam Generator Program Guidelines,'' performance criteria. The results of the Condition Monitoring Assessment demonstrated that all performance criteria were met during the FNP Unit 2 Fall 2002 refueling outage, and the results of the Operational Assessment show that all performance criteria will be met over the proposed operating period. Considering these actions, along with the improved SG design and reliability of Westinghouse replacement SGs, extending the SG tube inspection frequency does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change revises the SG inspection frequency requirements in TS 5.5.9.3.a for the FNP Unit 2 Spring 2004 refueling outage, to allow a 40[-]month inspection interval after one inspection, rather than after two consecutive inspections, with inspection results within the C–1 category.

The proposed change will not alter any plant design basis or postulated accident resulting from potential SG tube degradation. The scope of inspections performed during the last FNP Unit 2 refueling outage (*i.e.*, the first refueling outage following SG replacement) significantly exceeded the TS requirements for the scope of the first two refueling outages after SG replacement.

Primary-to-secondary leakage that may be experienced during all plant conditions is expected to remain within current accident analysis assumptions. The proposed change does not affect the design of the SGs, the method of SG operation, or reactor coolant chemistry controls. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. The proposed change involves a one-time extension to the SG tube inservice inspection frequency and therefore will not give rise to new failure modes. In addition, the proposed change does not impact any other plant systems or components.

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

The SG tubes are an integral part of the Reactor Coolant System (RCS) pressure boundary that are relied upon to maintain the RCS pressure and inventory. The SG tubes isolate the radioactive fission products in the reactor coolant from the secondary system. The safety function of the SGs is maintained by ensuring the integrity of the SG tubes. In addition, the SG tubes comprise the heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system.

SG tube integrity is a function of the design, environment, and current physical condition. Extending the SG tube inservice inspection frequency by one operating cycle will not alter the function or design of the SGs. SG inspections conducted during the first refueling outage following SG replacement demonstrated that the SGs do not have an active damage mechanism, and the scope of those inspections significantly exceeded the scope required by the TS. These inspection results were comparable to similar inspection results for second generation alloy 690 models of replacement SGs installed at other plants, and subsequent inspections at those plants yielded results that support this extension request. The improved design of the replacement SGs also provides reasonable assurance that significant tube degradation is not likely to occur over the proposed operating period.

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

*Attorney for licensee:* M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

*NRC Section Chief:* John A. Nakoski.

*Southern Nuclear Operating Company, Inc, Docket No. 50–364, Joseph M. Farley Nuclear Plant, Unit 2, Houston County, Alabama*

*Date of amendment request:* March 31, 2003.

*Description of amendment request:* The proposed amendments would modify Surveillance Requirement (SR) 3.4.11.1, for Farley, Unit 2 only by the addition of the following note that states, ''Not required to be performed for Unit 2 for the remainder of operating cycle 16 for Q2B31MOV800B.'' In addition, a temporary Technical Specification SR 3.4.11.4 is added to provide compensatory action for this block valve while SR 3.4.11.1 is suspended. Further, this SR requires that power to the Farley, Unit 2 Power Operated Relief Valve Q2B31MOV800B be checked at least every 24 hours for the remainder of operating cycle 16.

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

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

The proposed change to Surveillance Requirement (SR) 3.4.11.1 suspends the requirement to cycle test the Unit Two pressurizer power operated relief valve (PORV) block valve Q2B31MOV8000B for the remainder of operating cycle 16. This change will eliminate the remaining scheduled cycle tests for the PORV block valve during operating cycle 16. SR 3.4.11.4 is added to provide compensatory measures for verifying power available to the block valve at least every 24 hours. At the end of cycle 16, the proposed changes will no longer be in effect. Suspension of the cycle tests for the PORV block valve Q2B31MOV8000B may result in a small decrease in assurance that the block valve would cycle if required to isolate a stuck open PORV. However, experience with these valves has shown them to be very reliable and suspension of the remaining tests will not appreciably reduce reliability of the valve. There is no relationship between packing leakage on the PORV block valve and a postulated stuck open PORV. The proposed compensatory measure of verifying block valve power available on a 24 hour basis adds additional assurance that the block valve will close if demanded. Therefore, the probability of a previously evaluated accident remains acceptable is not significantly increased.

The proposed changes do not affect the consequences of a previously analyzed accident since the magnitude and duration of analyzed events are not impacted by this change. The dose consequences of the proposed change are bounded by LOCA [loss-of-coolant accident] analyses. Therefore, the consequences of a previously evaluated accident are unchanged.

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

The proposed changes involve no change to the physical plant. They allow for suspension of the PORV block valve Q2B31MOV8000B cycle tests for a limited time and provide for compensatory action to verify power to the PORV block valve. This valve provides an isolation function for a postulated stuck open or leaking pressurizer PORV. This condition is an analyzed event since it is bounded by the FNP [Farley Nuclear Plant] LOCA analyses. In addition to the isolation function, the block valve is required to remain open to allow the associated PORV to function automatically to control reactor coolant system (RCS) pressure. These changes do not impact the open function of the block valve since the normal position is open.

3. The proposed changes do not involve a significant reduction in a margin of safety.

The physical plant is unaffected by these changes. The proposed changes do not impact accident offsite dose, containment pressure or temperature, emergency core cooling system (ECCS) or reactor protection system (RPS) settings or any other parameter that could affect a margin of safety. The elimination of cycle testing of the PORV block valve Q2B31MOV8000B for the remainder of the Unit Two operating cycle and the addition of the proposed compensatory action that enhances assurance of valve operation are somewhat offsetting.

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

*Attorney for licensee:* M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

*NRC Section Chief:* John A. Nakoski.

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

*Date of amendment request:* February 26, 2003.

*Description of amendment request:* The proposed amendments would revise Technical Specifications Section 5.5.17, ''Containment Leakage Rate Testing Program,'' to reflect a one time deferral of the Type-A Containment Integrated Leak Rate Test (ILRT). The 10-year interval between ILRTs is to be extended to 15 years from the previous ILRTs that were completed in March 2002 for Unit 1 and March 1995 for Unit 2.

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

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

The proposed revision to Technical Specifications 5.5.17, ''Containment Leakage Rate Testing Program,'' involves a one-time extension to the current interval for Type A containment leak testing. The current test interval of ten (10) years would be extended on a one-time basis to no longer than fifteen (15) years from the last Type A test. The proposed Technical Specifications change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The reactor containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the reactor containment itself and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident.

The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications. Industry experience has shown, as documented in NUREG–1493 [''Performance-Based Containment Leak-Test Program''], that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. VEGP [Vogtle Electric Generating Plant] test history supports this conclusion. NUREG–1493 concluded, in part, that reducing the frequency of Type A containment leak tests to once per twenty (20) years leads to an imperceptible increase in risk. The integrity of the reactor containment is subject to two types of failure mechanism which can be categorized as (1) activity based and (2) time based. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as design change control and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the reactor containment itself combined with the containment inspections performed in accordance with ASME Section XI, the Maintenance Rule, and the containment coatings program serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing.

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

The proposed revision to Technical Specifications involves a one-time extension to the current interval for Type A containment leak testing. The reactor containment and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve the prevention or identification of any precursors of an accident. The proposed Technical Specifications change does not involve a physical change to the plant or the manner in which the plant is operated or controlled.

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

The proposed revision to Technical Specifications involves a one-time extension to the current interval for Type A containment leak testing. The proposed Technical Specifications change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The specific requirements and conditions of the Containment Leakage Rate Testing Program, as defined in Technical Specifications, exist to ensure that the degree of reactor containment structural integrity and leak tightness that is considered in the plant safety analysis is maintained. The overall containment leakage rate limit specified by Technical Specifications is maintained. The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications.

VEGP and industry experience strongly support the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with ASME Section XI, the Maintenance Rule, and the containment coatings program serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing.

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. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

*NRC Section Chief:* John A. Nakoski.

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

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

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

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

*Date of application for amendments:* April 8, 2003 as supplemented April 22, 2003.

*Brief description of amendments:* To revise, for one time only, a portion of Surveillance Requirement 3.5.2.3 of the Technical Specifications for the emergency core cooling system (ECCS). The revision will extend, until the refueling outage in the fall of 2003, the verification that the ECCS safety injection hot leg injection lines are full of water.

*Date of publication of individual notice in the **Federal Register:*** April 16, 2003 (68 FR 18712).

*Expiration date of individual notice:* May 1, 2003

## 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, 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 NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

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

*Date of application for amendment:* September 26, 2002.

*Brief description of amendment:* The amendment revises Technical Specification 3.9.1, ''Refueling Equipment Interlocks,'' to allow in-vessel fuel movement to continue if the refueling interlocks become inoperable. Specifically, the amendment adds Required Action A.2.1 to immediately block control rod withdrawal and Required Action A.2.2 to perform a verification that all of the control rods are fully inserted.

*Date of issuance:* April 28, 2003.

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

*Amendment No.:* 154.

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

Date of initial notice in **Federal Register:** November 26, 2002 (67 FR 70764).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 28, 2003.

*No significant hazards consideration comments received:* No.

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

*Date of application of amendments:* December 4, 2002.

*Brief description of amendments:* The amendments revised the Technical Specification 3.7.6 to require a minimum combined inventory of 155,000 gallons and remove the Condensate Storage Tank as a source of the combined inventory.

*Date of Issuance:* April 30, 2003.

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

*Amendment Nos.:* 330, 330 & 331.

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

*Date of initial notice in **Federal Register:*** January 31, 2003 (68 FR 2801).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 30, 2003.

*No significant hazards consideration comments received:* No

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

*Date of application for amendment:* December 6, 2002.

*Brief description of amendment:* The amendment increased the surveillance interval of the local power range monitor calibrations from 1000 megawatt-days/ton to 200 megawatt-days/ton.

*Date of issuance:* May 1, 2003.

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

*Amendment No.:* 277.

*Facility Operating License No. DPR–59:* Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 4, 2003 (68 FR 5674).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 1, 2003.

*No significant hazards consideration comments received:* No.

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

*Date of application for amendment:* August 16, 2002, as supplemented on March 26, April 16, and April 19, 2003.

*Brief description of amendment:* The amendment modified Technical Specification (TS) 3/4.10.A, ''Refueling Interlocks,'' and TS 3/4.10.D, ''Multiple Control Rod Removal,'' to provide an alternative required action if the refueling interlocks became inoperable during fuel movements in the reactor vessel. The amendment allowed fuel movements to continue in the reactor vessel should the refueling equipment interlocks become inoperable.

*Date of issuance:* April 21, 2003.

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

*Amendment No.:* 199.

*Facility Operating License No. DPR–35:* Amendment revised the TSs.

*Date of initial notice in **Federal Register:*** December 10, 2002 (67 FR 75872).

The March 26, April 16, and April 19, 2003, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 21, 2003.

*No significant hazards consideration comments received:* No.

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

*Date of application for amendment:* January 23, 2003, as supplemented February 24, and April 17, 2003.

*Brief description of amendment:* This amendment modifies the Pilgrim Nuclear Power Station Technical Specification (TS) requirements for the Emergency Core Cooling System (ECCS) during shutdown conditions. The amendment changes the Core Spray and Low Pressure Coolant Injection System's TS requirements to be applicable during the Run, Startup, and Hot Shutdown Modes. The amendment also modifies the High Drywell Pressure Instrumentation TSs to require the instrumentation to be Operable during the Run, Startup and Hot Shutdown Modes. Unnecessary TS requirements are removed based on the plant's operating Mode. Other changes are administrative in nature.

*Date of issuance:* April 22, 2003.

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

*Amendment No.:* 200.

*Facility Operating License No. DPR–35:* Amendment revised the TSs.

*Date of initial notice in **Federal Register:*** March 18, 2003 (68 FR 12952).

The supplements dated February 24, and April 17, 2003, provided additional information that clarified the application, and did not expand the scope of the application or change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 22, 2003.

*No significant hazards consideration comments received:* No.

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

*Date of application for amendment:* September 18, 2002.

*Brief description of amendment:* The amendment revises several Technical Specifications Limiting Conditions for Operations and Administrative sections to correct or clarify certain requirements and information.

*Date of issuance:* April 23, 2003.

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

*Amendment No:* 157.

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

*Date of initial notice in **Federal Register:*** December 10, 2002 (67 FR 75871).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 23, 2003.

*No significant hazards consideration comments received:* No.

*Exelon Generation Company, LLC, Docket Nos. 50–237, Dresden Nuclear Power Station, Unit 2, Grundy County, Illinois*

*Date of application for amendment:* January 31, 2003, as supplemented March 7, 2003.

*Brief description of amendment:* The amendment revises the safety limit minimum critical power ratio for Unit 2 for two loop operation and for single loop operation.

*Date of issuance:* April 22, 2003.

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

*Amendment No.:* 199.

*Facility Operating License No. DPR–19:* The amendment revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** March 4, 2003 (68 FR 10279). The supplement dated March 7, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 22, 2003.

*No significant hazards consideration comments received:* No.

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

*Date of application for amendments:* April 10, 2002, as supplemented March 10, 2003.

*Brief description of amendments:* The amendments revised the Technical Specifications to relocate emergency diesel generator maintenance inspection requirements from Section 4.8.1.1.2.e.1 to the Technical Requirements Manual.

*Date of issuance:* April 18, 2003.

*Effective date:* As of the date of issuance and shall include the relocation of the emergency diesel generator maintenance requirements of Technical Specification 4.8.1.1.2.e.1 to the Technical Requirements Manual within 30 days.

*Amendment Nos.:* 165 and 128.

*Facility Operating License Nos. NPF–39 and NPF–85:* The amendments revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** May 28, 2002 (67 FR 36926). The supplement dated March 10, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 18, 2003.

*No significant hazards consideration comments received:* No.

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

*Date of application for amendments:* November 27, 2002.

*Brief description of amendments:* These amendments deleted TS 6.8.4.c, ''Post-accident Sampling,'' and thereby eliminated the requirements to have and maintain the post accident sampling system for Limerick Generating Station, Units 1 and 2. The amendments also addressed related changes to TS 6.8.4.a, ''Primary Coolant Sources Outside Containment.''

*Date of issuance:* April 25, 2003.

*Effective date:* As of date of issuance and shall be implemented within 180 days.

*Amendment Nos.:* 166 and 129.

*Facility Operating License Nos. NPF–39 and NPF–85.* The amendments revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** January 21, 2003 (68 FR 2802).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 25, 2003.

*No significant hazards consideration comments received:* No.

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

*Date of application for amendment:* June 4, 2002.

*Brief description of amendment:* This amendment revises the pressure temperature limits for 22- and 32-effective full power years for Perry Nuclear Power Plant. The June 4, 2002, application also contained a request for exemption from applying Appendix G of the 1995 American Society of Mechanical Engineers Boiler and Pressure Vessel Code and approval for using Code Case N–640, which permits the use of the plain strain fracture toughness (KIc) curve instead of the crack arrest fracture toughness (KIa) curve for reactor pressure vessel materials in determining the P–T limits.

*Date of issuance:* April 29, 2003.

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

*Amendment No.:* 127.

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

*Date of initial notice in* ***Federal Register:*** December 10, 2002 (67 FR 75878).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 29, 2003.

*No significant hazards consideration comments received:* No.

*Florida Power and Light Company, Docket No. 50–389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida*

*Date of amendment request:* October 15, 2002, as supplemented February 28, 2003.

*Description of amendment request:* The amendment modifies the reactor coolant system flow rate from 363,000 gallons per minute (gpm) to 355,000 gpm in Technical Specifications (TSs) Table 3.3–2 and in a footnote for Table 2.2–1.

*Date of Issuance:* April 18, 2003.

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

*Amendment No.:* 131.

*Facility Operating License No. NPF–16:* Amendment revised the TSs.

*Date of initial notice in* ***Federal Register:*** November 12, 2002 (67 FR 68737).

The February 28, 2003, supplement did not affect the original proposed no significant hazards determination, or expand the scope of the request as noticed in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 18, 2003.

*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:* July 23, 2002.

*Brief description of amendments:* The amendments revise certain 18-month surveillance requirements by eliminating the condition that testing be conducted ''during shutdown,'' or ''during the COLD SHUTDOWN or REFUELING MODE'' (*i.e.,* shutdown conditions).

*Date of issuance:* April 22, 2003.

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

*Amendment Nos.:* 275 and 257.

*Facility Operating License Nos. DPR–58 and DPR–74:* Amendments revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** September 17, 2002 (67 FR 58647).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 22, 2003.

*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:* January 14, 2003.

*Brief description of amendments:* The amendments modify Technical Specification (TS) 3.7.5.1 to add an exception to Limiting Condition for Operation 3.0.4 for the control room emergency ventilation system (CREVS). This exception allows movement of irradiated fuel assemblies to begin while one of the two CREVS pressurization trains is inoperable, provided the appropriate TS action requirements are implemented. The amendments are consistent with the standard TSs for Westinghouse plants (NUREG 1431, Revision 2, ''Standard Technical Specifications, Westinghouse Plants,'' dated April 30, 2001).

*Date of issuance:* April 25, 2003.

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

*Amendment Nos.:* 276 and 258.

*Facility Operating License Nos. DPR–58 and DPR–74:* Amendments revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** March 4, 2003 (68 FR 10280).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 25, 2003.

*No significant hazards consideration comments received:* No.

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

*Date of application for amendment:* October 26, 2001, as supplemented by letters dated June 7 and November 22, 2002.

*Brief description of amendment:* The amendment revised Section 6.0, ''Administrative Controls,'' of the Technical Specifications (TSs) to clarify and relocate existing requirements, make wording improvements, and make the TSs consistent with the Unit 2 TSs. The revised Section 6.0 is consistent with the ''Standard Technical Specifications for General Electric plants, BWR [Boiling Water Reactor]/4'' (NUREG–1433, Revision 2).

*Date of issuance:* April 23, 2003.

*Effective date:* April 23, 2003, to be implemented within 90 days of issuance.

*Amendment No.:* 181.

*Facility Operating License No. DPR–63:* Amendment revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** January 8, 2002 (67 FR 928).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 23, 2003.

*No significant hazards consideration comments received:* No.

*Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin*

*Date of application for amendment:* July 26, 2002, as supplemented February 27, March 14, March 19, March 21 (2 letters), and April 3, 2003.

*Brief description of amendment:* The amendment revises technical specifications for use of Westinghouse 422 VANTAGE + nuclear fuel with PERFORMANCE + features.

*Date of issuance:* April 4, 2003.

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

*Amendment No.:* 167.

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

*Date of initial notice in* ***Federal Register:*** September 3, 2002 (67 FR 56322).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 4, 2003.

*No significant hazards consideration comments received:* No.

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

*Date of application for amendment:* September 19, 2002, as supplemented February 28, 2003.

*Brief description of amendment:* The amendment relocates TS Surveillance Requirement (SR) 4.6.B.2, ''Reactor Vessel Temperature and Pressure,'' and the associated TS Bases to Section 4.2 of the Updated Safety Analysis Report. It also implements the Boiling Water Reactor Vessel and Internals Project reactor pressure vessel integrated surveillance program at Monticello and demonstrates compliance with the requirements of Title 10 of the Code of Federal Regulations, Part 50, Appendix H, ''Reactor Vessel Material Surveillance Program Requirements.''

*Date of issuance:* April 22, 2003

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

*Amendment No.:* 135.

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

*Date of initial notice in* ***Federal Register:*** October 29, 2002 (67 FR 66012).

The supplement of February 28, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 22, 2003.

*No significant hazards consideration comments received:* No.

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

*Date of amendment request:* October 8, 2002, and its supplements dated December 3, 2002, and March 4, 2003.

*Brief description of amendment:* The amendment modifies Technical Specification 2.3.a, ''Emergency Core Cooling System,'' to extend *the* allowed outage time for a single low pressure safety injection pump from the existing 24 hours to 7 days. In addition, the word ''pump'' has been replaced with the word ''train.''

*Date of issuance:* April 29, 2003

*Effective date:* April 29, 2003, and shall be implemented within 60 days from the date of issuance.

*Amendment No.:* 217.

*Facility Operating License No. DPR–40:* Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 12, 2002 (67 FR 68740).

The supplemental letters dated December 3, 2002, and March 4, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed or revise the proposed technical specification changes and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 29, 2003.

*No significant hazards consideration comments received:* No.

*Saxton Nuclear Experimental Corporation (SNEC) and GPU Nuclear, Inc., Docket No. 50–146 Saxton Nuclear Experimental Facility (SNEF), Bedford County, Pennsylvania*

*Date of application for amendment:* February 2, 2000, as supplemented on

June 23, August 11, September 18 and December 4, 2000; January 30, February 14, March 15 and 19, June 20, July 2 and September 4, 2001; and January 11 and 24, February 4, May 22 and 28, July 11, August 20, September 17, 23, 24, and 26, October 10, and December 16, 2002.

*Brief description of amendment:* The amendment revises Amended Facility License No. DPR–4 for the SNEF to annotate approval of the SNEF License Termination Plan.

*Date of issuance:* March 28, 2003.

*Effective date:* Date of issuance to be implemented no later than 30 days from the date of issuance.

*Amendment No.:* 18.

*Amended Facility License No. DPR–4:* Amendment added a new license condition to require the licensees to implement and maintain in effect all provisions of the approved SNEF License Termination Plan.

*Date of initial notice in **Federal Register:*** November 29, 2000.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 28, 2003.

*No significant hazards consideration comments received:* No.

*STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas*

*Date of amendments request:* November 5, 2001, as supplemented by letters dated October 23, 2002, and January 15, 2003.

*Brief description of amendments:* The proposed amendments convert the current Technical Specification (TS) Section 6.0 of the STP, Units 1 and 2, TS to the Improved Technical Specifications based on NUREG–1431, ''Standard Technical Specification for Westinghouse Plants.''

*Date of issuance:* April 24, 2003.

*Effective date:* As of its date of issuance and shall be implemented within 6 months from the date of issuance.

*Amendment Nos.:* Unit 1–151; Unit 2–139.

*Facility Operating License Nos. NPF–76 and NPF–80:* The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 5, 2002 (67 FR 5335).

The October 23, 2002, and January 15, 2003, supplemental letters provided clarifying information that was within the scope of the original **Federal Register** notice (67 FR 5335) and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 24, 2003.

*No significant hazards consideration comments received:* No.

*Tennessee Valley Authority, Docket No. 50–327, Sequoyah Nuclear Plant, Unit 1, Hamilton County, Tennessee*

*Date of application for amendment:* February 28, 2003.

*Brief description of amendment:* The amendment approves the use of an alternate methodology using a through-bolted connection frame to restore the steam generator (SG) compartment roof after replacement of the SGs, and a revision of the Updated Safety Analysis Report (UFSAR) to reflect the approval of the methodology.

*Date of issuance:* April 25, 2003.

*Effective date:* As of the date of issuance, to be incorporated into the UFSAR at the time of its next update.

*Amendment No.:* 184.

*Facility Operating License No. DPR–77:* Amendment revises the UFSAR.

*Date of initial notice in **Federal Register:*** March 14, 2003 (68 FR 12382).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 25, 2003.

*No significant hazards consideration comments received:* No

*Tennessee Valley Authority, Docket No. 50–327, Sequoyah Nuclear Plant (SQN), Unit 1, Hamilton County, Tennessee*

*Date of application for amendment:* February 28, 2003.

*Description of amendment:* The amendment approves a revision of the SQN Updated Final Safety Analysis (UFSAR) to include a change to the methodology for connecting reinforcing steel bars during restoration of the Unit 1 concrete shield building dome as part of the steam generator replacement project. This modification to the shield building concrete dome is necessary to support removal of the original steam generators and installation of the replacement steam generators.

*Date of issuance:* April 24, 2003.

*Effective date:* As of the date of issuance to be incorporated into the UFSAR at the time of its next update.

*Amendment No.:* 283.

*Facility Operating License No. DPR–77:* Amendment revises the Operating License.

*Date of initial notice in* **Federal Register**: March 17, 2003 (68 FR 12718).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 24, 2003.

*No significant hazards consideration comments received:* No.

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

*Date of application for amendment:* April 8, 2003, as supplemented April 22, 2003.

*Brief description of amendment:* The amendment revises, for one time only, a portion of Surveillance Requirement (SR) 3.5.2.3 of the Watts Bar Technical Specifications for the emergency core cooling system (ECCS). The revision extends, until the refueling outage in the fall of 2003, the verification that the ECCS safety injection hot leg injection lines are full of water. SR 3.5.2.3 currently requires a verification frequency of 31 days.

*Date of issuance:* May 1, 2003.

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

*Amendment No.:* 43.

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

*Public comments requested as to proposed no significant hazards consideration:* Yes. (68 FR 18712 dated April 16, 2001). That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by May 16, 2003, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendment. The April 22, 2003, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original request.

The Commission's related evaluation of the amendments, finding of exigent circumstances, and final no significant hazards consideration determination are contained in a Safety Evaluation dated May 1, 2003.

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

*NRC Section Chief:* Allen G. Howe.

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

*Date of application for amendment:* October 3, 2002.

*Brief description of amendment:* The amendment revises Limiting Condition for Operation 3.1.8, ''Physics Tests Exceptions—Mode 2,'' to reduce the required number of channels from four to three channels for certain functions in Table 3.3.1–1, ''Reactor Trip System Instrumentation.''

*Date of issuance:* April 21, 2003.

*Effective date:* April 21, 2003, and shall be implemented within 60 days of the date of issuance.

*Amendment No.:* 154.

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

*Date of initial notice in* ***Federal Register:*** November 26, 2002 (67 FR 70771).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 21, 2003.

No significant hazards consideration comments received. No.

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

Date of amendment request: October 1, 2002.

*Brief description of amendment:* The amendment revises Limiting Condition for Operation 3.1.8, ''Physics Tests Exceptions—Mode 2,'' to reduce the required number of channels from four to three channels for certain functions in Table 3.3.1–1, ''Reactor Trip System Instrumentation.''

*Date of issuance:* April 21, 2003.

*Effective date:* April 21, 2003, and shall be implemented within 90 days of the date of issuance.

*Amendment No.:* 151.

*Facility Operating License No. NPF–42.* The amendment revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** November 12, 2002 (67 FR 68746).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 21, 2003.

*No significant hazards consideration comments received:* No.

*Yankee Atomic Electric Co., Docket No. 50–29, Yankee Nuclear Power Station (YNPS) Franklin County, Massachusetts*

*Brief description of amendment:* The amendment revises the YNPS License and Technical Specifications to delete operational and administrative requirements that would no longer be required once the spent nuclear fuel has been transferred from the spent fuel pool to the Independent Spent Fuel Storage Installation.

*Date of issuance:* April 17, 2003.

*Effective date:* April 17, 2003.

*Amendment No.:* 157.

*Facility Operating License No. DPR–3.* Amendment revises the License and Technical Specifications.

*Date of initial notice in* ***Federal Register:*** February 18, 2003 (68 FR 7823).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 17, 2003.

*No significant hazards consideration comments received:* No.

Dated at Rockville, Maryland, this 5th day of May 2003.

For the Nuclear Regulatory Commission.

**John A. Zwolinski,**

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

[FR Doc. 03–11697 Filed 5–12–03; 8:45 am]

**BILLING CODE 7590–01–P**

## NUCLEAR REGULATORY COMMISSION

## Notice of Availability of Model Application Concerning Technical Specification Improvement To Eliminate Post Accident Sampling Requirements for Babcock and Wilcox Reactors Using the Consolidated Line Item Improvement Process

**AGENCY:** Nuclear Regulatory Commission.

**ACTION:** Notice of availability.

**SUMMARY:** Notice is hereby given that the staff of the Nuclear Regulatory Commission (NRC) has prepared a model application relating to the elimination of post accident sampling requirements for Babcock and Wilcox (B&W) Reactors. The purpose of this model is to permit the NRC to efficiently process amendments that propose to remove requirements for Post Accident Sampling Systems (PASS) from Technical Specifications (TS). Licensees of nuclear power reactors to which the model applies may request amendments utilizing the model application.

**DATES:** The NRC staff issued a **Federal Register** Notice (68 FR 10052, March 3, 2003) which provided a model safety evaluation (SE) and a model no significant hazards consideration (NSHC) determination relating to elimination of requirements for PASS for B&W Reactors. The NRC staff hereby announces that the model SE and NSHC determination may be referenced in plant-specific applications to eliminate requirements for post accident sampling. The staff has posted a model application on the NRC web site to assist licensees in using the consolidated line item improvement process (CLIIP) to eliminate PASS-related TS. The NRC staff can most efficiently consider applications based upon the model application if the application is submitted within a year of this **Federal Register** Notice.

**FOR FURTHER INFORMATION CONTACT:** Robert Dennig, Mail Stop: O–12H4, Division of Regulatory Improvement Programs, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone 301–415–1156.

**SUPPLEMENTARY INFORMATION:**

### Background

Regulatory Issue Summary 2000–06, ''Consolidated Line Item Improvement Process for Adopting Standard Technical Specification Changes for Power Reactors,'' was issued on March 20, 2000. The CLIIP is intended to improve the efficiency of NRC licensing processes. This is accomplished by processing proposed changes to the standard technical specifications (STS) in a manner that supports subsequent license amendment applications. The CLIIP includes an opportunity for the public to comment on proposed changes to the STS following a preliminary assessment by the NRC staff and finding that the change will likely be offered for adoption by licensees. The CLIIP directs the NRC staff to evaluate any comments received for a proposed change to the STS and to either reconsider the change or to proceed with announcing the availability of the change for proposed adoption by licensees. Those licensees opting to apply for the subject change to TS are responsible for reviewing the staff's evaluation, referencing the applicable technical justifications, and providing any necessary plant-specific information. Each amendment application made in response to the notice of availability will be processed and noticed in accordance with applicable rules and NRC procedures.

This notice involves the elimination of requirements for PASS and related administrative controls in TS for B&W Reactors. This proposed change was proposed for incorporation into the STS by the B&W Owners Group (BWOG) participants in the Technical Specification Task Force (TSTF) and is designated TSTF–442. TSTF–442 is supported by the NRC staff's SE dated November 14, 2002 (ADAMS Accession Number ML0225601190), for the BWOG topical report BAW–2387, ''Justification for the Elimination of the Post Accident Sampling System (PASS) from the Licensing Basis of Babcock and Wilcox Plants,'' which was submitted to the NRC on June 25, 2001. The BWOG request followed the staff's approval of similar requests for elimination of PASS requirements from the Combustion Engineering Owners Group (CEOG), the