Sidell, Permit No. 2003–017; H. William Detrich, Permit No. 2003–018.

Nadene G. Kennedy,

Permit Officer.

[FR Doc. 03–6468 Filed 3–17–03; 8:45 am]

BILLING CODE 7555–01–M

NUCLEAR REGULATORY COMMISSION

Meetings; Sunshine Act

DATE: Weeks of March 17, 24, 31, April 7, 14, 21, 2003.

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

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of March 17, 2003

Thursday, March 20, 2003

10 a.m. Briefing on status of Office of Nuclear Security and Incident Response (NSIR) Programs, Performance, and Plans (closed—Ex. 1).

2 p.m. Discussion of Management Issues (closed—Ex. 2).

Week of March 24, 2003—Tentative

Thursday, March 27, 2003

10 a.m. Briefing on status of Office of Nuclear Regulatory Research (RES) Programs, Performance, and Plans.

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

Week of March 31, 2003—Tentative

There are no meetings scheduled for the week of March 31, 2003.

Week of April 7, 2003—Tentative

Friday, April 11, 2003

9 a.m. Meeting with Advisory Committee on Reactor Safeguards (ACRS) (public meeting) (contact: John Larkins, 301–415–7360).

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

12:30 p.m. Discussion of Management Issues (closed—Ex. 2).

Week of April 14, 2003—Tentative

There are no meetings scheduled for the week of April 14, 2003.

Week of April 21, 2003—Tentative

There are no meetings scheduled for the week of April 21, 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 March 6, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Discussion of Legislative Issues (Closed—Ex. 9)'' be held on March 6, and on less than one week's notice to the public.

By a vote of 5–0 on March 6, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Affirmation of Final Rule to Standardize the Process for Allowing a Licensee to Release Part of Its Reactor Facility or site for Unrestricted Use Before NRC Has Approved Its License Termination Plan'' be held on March 7, and on less than one week's notice to the public.

By a vote of 5–0 on march 7, the Commission determined pursuant to U.S.C. 552b(E) and § 9.107(a) of the Commission's rules that ''Discussion of legislative Issues (Closed—Ed. 9)'' be held on March 7, and on less than one week's notice to the public.

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The NRC Commission meeting schedule can be found on the Internet at: *http://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 220555 (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: March 13, 2003.

David Louis Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 03–6546 Filed 3–14–03; 11:47 am]

BILLING CODE 4590–01–M

NUCLEAR REGULATORY COMMISSION

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

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, February 21, 2003, through March 6, 2003. The last biweekly notice was published on March 4, 2003 (68 FR 10277).

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 April 17, 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,[1] 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).

[1] The most recent version of Title 10 of the Code of Federal Regulations, published January 1, 2002, inadvertently omitted the last sentence of 10 CFR 2.714 (d) and paragraphs (d)(1) and (d)(2) regarding petitions to intervene and contentions. For the complete, corrected text of 10 CFR 2.714 (d), please *see* 67 FR 20884; April 29, 2002.

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. 50–289, Three Mile Island Nuclear Station, Unit 1 (TMI–1), Dauphin County, Pennsylvania

Date of amendment request: January 16, 2003

Description of amendment request: The proposed amendment would revise the TMI–1 Technical Specifications to incorporate changes associated with the Cycle 15 core reload design analysis.

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

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

Response: No.

The proposed Technical Specification limits (Figure 2.1–1) and reactor protection system (RPS) trip setpoints (Table 2.3–1) are developed in accordance with the methods and assumptions described in NRC-[Nuclear Regulatory Commission] approved Framatome ANP Topical Reports BAW–10179 P–A, ''Safety Criteria and Methodology for Acceptable Cycle Reload Analyses'' and BAW–10187 P–A, ''Statistical Core Design for B&W-[Babcock&Wilcox-] Designed 177 FA Plants.'' The core thermal-hydraulic code (LYNXT) and CHF [critical heat flux] correlation (BWC) have been approved for use with these methods and the Mark-B fuel type utilized at TMI Unit 1. The proposed Technical Specification requirements on Variable Low Pressure Trip (VLPT) instrument operating conditions (Table 3.5–1) and surveillances (Table 4.1–1) are consistent with the VLPT requirements that were last contained in the TMI Unit 1 Technical Specifications prior to Cycle 7. The existing flux-flow trip setpoint and power/pump monitor trip have been shown to provide adequate DNB [departure from nucleate boiling] protection for Updated Final Safety Analysis Report (UFSAR) DNB-limiting loss of coolant events.

The margin retained for penalties such as transition core effects, by imposing a Thermal Design Limit of 1.40 in all DNB analyses supporting the proposed change, has been shown to be sufficient to offset the current mixed core conditions at TMI Unit 1, where the Mark-B12 fuel design with fine mesh debris filter is co-resident with earlier, non-debris filter Mark-B fuel designs. Therefore the previous commitment to require a higher minimum RCS [reactor coolant system] flow (105.5% of design flow instead of 104.5%) to offset transition core penalties is no longer necessary.

Reload cycles are designed and operated with maximum steady-state radial-local peaking factors that are bounded by UFSAR assumptions used to determine the dose consequences from fuel handling accidents.

The proposed change to Technical Specification 3.5.2.2.a is only an administrative correction.

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 Technical Specification limits (Figure 2.1–1) and reactor protection system (RPS) trip setpoints (Table 2.3–1) provide core protection safety limits and Variable Low Pressure Trip setpoints developed in accordance with NRC-approved methods and assumptions. The transition core penalty resulting from Mark-B12 fuel with fine mesh debris filters co-residing with earlier, non debris filter Mark-B fuel has been demonstrated to be sufficiently bounded by the analyses supporting the proposed amendment. Therefore the previous commitment to require a higher minimum RCS flow (105.5% of design flow instead of 104.5%) to offset transition core penalties is no longer necessary. These changes have been evaluated for their impact on the design and operation of plant structures, systems, and components. These changes do not introduce any new accident precursors and do not involve any alterations to plant configurations, which could initiate a new or different kind of accident.

The proposed change to Technical Specification 3.5.2.2.a is only an administrative correction.

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

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

Response: No.

The proposed reactor protection system (RPS) trip setpoints (Table 2.3–1) ensure core protection safety limits will be preserved during power operation. The proposed safety limits and setpoints are developed in accordance with NRC-approved methods and assumptions. The margin retained for penalties such as transition core effects, by imposing a Thermal Design Limit of 1.40 in all DNB analyses supporting the proposed change, has been shown to be sufficient to offset the current mixed core conditions at TMI Unit 1. The margin available between minimum DNBR [departure from nucleate boiling ratio] results for UFSAR loss of coolant flow events and the Thermal Design Limit of 1.40 is significant and is similar to DNB margin results for the current non-SCD [Statistical Core Design] analysis.

Reload cycles are designed and operated with maximum steady-state radial-local peaking factors that are bounded by UFSAR assumptions used to determine the dose consequences from fuel handling accidents.

The proposed change to Technical Specification 3.5.2.2.a is only an administrative correction.

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: Edward J. Cullen, Jr., Esquire, Vice President, General Counsel and Secretary, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: Richard J. Laufer.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: February 14, 2003.

Description of amendment request: The amendment would allow an increase in the maximum decay heat of spent fuel stored in Spent Fuel Pools (SFPs) C and D from 1.0 MBTU/hr to 7.0 MBTU/hr in Technical Specification 5.6.3.d. The amendment would also increase the allowable SFP temperatures from 140 degrees F to 150 degrees F under normal and emergency conditions other than a design-basis Loss-of-Coolant Accident (LOCA). For a LOCA, the maximum allowed SFP temperature would increase from 150 degrees F to 160 degrees F.

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

A written evaluation of the significant hazards consideration of a proposed license amendment is required by 10 CFR 50.92. Progress Energy Carolinas, Inc. (alternately known as Carolina Power & Light Company) has evaluated the proposed amendment and determined that it involves no significant hazards consideration. According to 10 CFR 50.92, a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; 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 determination is as follows:

Proposed Change

The change involves an increase in the maximum decay heat of spent fuel stored in Spent Fuel Pools (SFPs) C and D from 1.0 MBTU/hr to 7.0 MBTU/hr, and an increase in the allowable SFP temperatures.

Basis

This change does not involve a significant hazards consideration for the following reasons:

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

The license amendment only increases the heat load from the Fuel Pool Cooling and Cleanup System (FPCCS) and the maximum allowable pool temperature. The changes do not modify the design of Structures, Systems and Components (SSCs) that could initiate an accident. The FHB [Fuel Handling Building] Emergency Exhaust System mitigates the consequences of a fuel handling accident in the Fuel Handling Building. This system has been evaluated for the conditions that would exist with the higher SFP temperatures and it was found that there would be no decrease in the charcoal efficiency. As a result, there was no increase in the doses from the fuel handling accident in the FHB. Therefore, the change does not result in any increase in the probability or consequences in any accident previously analyzed.

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

The increase in the SFP decay heat load and the SFP temperature limit does not involve new plant components or procedures. No significant impact on any postulated accident is made due to this change since the required cooling capacity is maintained to the SFPs and the FPCCS, and the SFPs will operate within design parameters.

For the activation of SFPs C and D, Progress Energy Carolinas, Inc. performed a Probabilistic Safety Analysis (PSA) of a total loss of SFP forced cooling. That analysis concluded that the probability of spent fuel rack uncovery was not credible. That analysis remains bounding for this license amendment application.

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

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

The proposed changes do not affect the design or operation of the barriers to fission product release (fuel cladding, reactor coolant system pressure boundary, and containment boundary). The change in the SFPs C and D decay heat load is bounded by the heat load used in the analysis of the safety-related systems for design basis accidents. Therefore, there is no impact in the margin of safety.

Based on these considerations, the proposed change does not involve a significant reduction on 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: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Allen Howe.

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

Date of amendment request: February 19, 2003.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 5.5.10, ''Steam Generator (SGs) Tube Surveillance Program.'' The proposed amendments would relocate to TS 5.5.21 the TS 5.5.10 program requirements that apply to the original SGs and would provide a new TS 5.5.10 that contains program requirements that would apply to the new SGs when they are installed.

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:

Pursuant to 10 CFR 50.91, Duke has made the determination that this amendment request does not involves a significant hazard by applying the three standards established by the NRC regulations in 10 CFR 50.92 as described below.

First Standard

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

The proposed amendment will revise Technical Specification (TS) 5.5.10 to delete and clarify replacement steam generator (SG) surveillance requirements applicable to the replacement of the SGs following their installation. The proposed amendment does not result in any changes to the design or methods of operation of the facility or any of its structures, systems or components (SSC). The SG repair methods that would be deleted are not applicable to the replacement SGs due to the use of improved materials and design. Defects found during future replacement SG tube inspections that exceed the limits in the new TS 5.5.10 will be removed from service by plugging rather than being repaired. The accident analyses and assumptions made in the Updated Final Safety Analysis Report (UFSAR) Chapter 15, Accident Analyses, are not changed as a result of the proposed changes. There are no changes resulting from the new TS 5.5.10 that could affect the function of preventing or mitigating any of these accidents. The proposed change does not increase the likelihood of the malfunction of an SSC that may increase the probability or consequences of an accident. The relocated surveillance requirements for the current steam generators will not change as a result of the proposed TS changes. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Second Standard

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

The proposed changes to the SG tube surveillance TS will delete or modify surveillance requirements that would otherwise not be applicable to the replacement steam generators. SG Tubes found to exceed the plugging limit criteria of TS 5.5.10 for continued service will be removed from service by plugging rather than being repaired. The plugging limit is unchanged by the proposed amendment. These changes will not introduce any adverse changes to the facilities' design bases or postulated accidents resulting from potential tube degradation. The proposed amendment does not affect the design of SGs, their method of operation, or primary coolant chemistry controls. In addition, the proposed amendment does not impact any other SSC. Surveillance requirements for the current SGs will not change prior to their removal from service as a result of the proposed changes. Therefore, the proposed changes do not create the possibility of a new or different type of accident from any accident previously evaluated.

Third Standard

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

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. These barriers are unaffected by the changes proposed in this LAR. The steam generator tubes are an integral part of the reactor coolant pressure boundary. Repairing SG tubes by previously approved methods of sleeving or rerolling are considered to be an equivalent boundary to plugging a steam generator tube as has also been previously approved. Therefore, the margin of safety is not reduced by the changes proposed in this license amendment request.

Conclusion

Based upon the proceeding evaluation, performed pursuant to 10 CFR 50.92, Duke Energy Corporation has concluded that approval and implementation of this license amendment request at the Oconee Nuclear Station will not involve a significant hazards consideration. The proposed changes revise the steam generator surveillance requirements to be consistent with the replacement steam generators. Following implementation of the changes proposed in this license amendment request, the Oconee steam generators will continue to be operated in a safe and conservative manner.

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: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: January 29, 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 proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

Response: No.

Most accident conditions will not result in an increase in K-effective (K_{eff}) of the fuel stored in the rack. However, there are accidents that can be postulated to increase reactivity. 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. Therefore, 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.

A vertical drop accident condition directly upon a cell will cause damage to the racks in the active fuel region. The proposed >2000 ppm [parts per million] TS [technical specification] limit will insure that K_{eff} does not exceed 0.95. A fuel assembly dropped on top of the rack will not deform the rack structure such that criticality assumptions are invalidated. The rack structure is such that [after rack deformation] an assembly positioned horizontally on top of the rack is more than eight inches away from the upper end of the active fuel region of the stored assemblies. This distance precludes 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 of 825 ppm. The distance between all the rack modules and the pool walls is [nominally] less than the width of a fuel assembly.

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 [Uranium-235] loading and burnup. The presence of boron in the pool water assumed in the analysis has been shown to substantially offset the worst case reactivity effect of a misplaced fuel assembly for any configuration. The boron requirement of 825 ppm is less than the proposed >2000 ppm minimum boron TS limit. Therefore, 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 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).

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, a new criticality analysis was performed. The analysis 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 proposed ANO–2 [Arkansas Nuclear One, Unit 2] TS minimum concentration of >2000 ppm. Therefore, 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 that 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 [spent fuel pool] storage racks are 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 subcritical margin of five percent. This has been verified by criticality analyses.

Credit for soluble boron in the SFP water is permitted under accident conditions. The proposed change 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 that the maximum K_{eff} does not exceed 0.95. By increasing the minimum boron concentration to >2000 ppm, 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. All structural requirements were shown to be satisfied, so all 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 Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: December 17, 2002.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 5.5.10, ''Ventilation Filter Testing Program,'' to adopt the requirements of the American Society for Testing and Materials Standard (ASTM) D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon.'' The proposed TS revisions are in response to Nuclear Regulatory Commission (NRC) Generic Letter (GL) 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.'' The NRC had previously published a notice of consideration on December 12, 2001 (66 FR 64292) regarding a similar proposal from the licensee in response to GL 99–02. However, in response to a request for additional information from the NRC dated March 29, 2002, the licensee has now revised its proposed amendment. In addition to withdrawing the prior request to change the maximum control room ventilation system (CRVS) differential pressure in TS 5.5.10.d, the proposed amendment would revise the TSs: (1) To provide a CRVS methyl iodide removal efficiency of greater than or equal to 95.5% and remove the notation that there is a 1-inch charcoal bed depth; (2) to allow for the continued use of the existing CRVS through Refueling Outage 13, in order to design, fabricate, and install a 2-inch charcoal filter bed; (3) to add a note in the TS requiring a demonstration of charcoal efficiency of 93% when changing the charcoal in the existing CRVS bed prior to any fuel movement in the upcoming Refueling Outage 12 and every 6 months thereafter until the new beds are installed. The proposed amendment also seeks an exception from the factor of safety of two for the Containment Fan Cooler Units due to the plant's design.

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

Response: The proposed license amendment adopts the new test method and acceptance criteria of ASTM D3803–1989 for activated charcoal filters. The changes require laboratory performance testing of adsorber carbon that yields a more accurate result than the testing currently required by the TS. The proposed change to delete non-conservative TS requirements for testing of adsorber carbon is not a plant accident initiator as described in the Final Safety Analysis Report (FSAR). The proposed amendment does not change the function of any structure, system or component (SSC). The function of the ventilation systems is filtration of radiological releases during postulated accidents. The proposed changes will provide greater assurance that this function is provided. The revised TS requirements are for laboratory tests that are currently in place to address Generic Letter 99–02, with one exception to the safety factor of 2, and accommodate the change of the Control Room Ventilation System (CRVS) charcoal beds to two inches. The change only affects the TS testing requirements since the modification to the CRVS will be accomplished separately from the TS change. The TS changes will not result in any changes to the efficiency assumed in accident analysis. The changes do not alter, degrade or prevent actions described or assumed in an accident described in the FSAR. Therefore, the proposed amendment does not change the possibility of an accident previously evaluated or significantly increase the consequences of an accident previously evaluated.

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

Response: The proposed license amendment adopts the new test method and acceptance criteria of ASTM D3803–1989 for activated charcoal filters. The change does not involve any modifications to the plant but will accommodate the planned modification of the CRVS to change the charcoal beds from 1 inch to 2 inches. The change will not require changes to how the plant is operated nor will it affect the operation of the plant. The changes require laboratory performance testing of adsorber carbon that yields a more accurate result than the testing currently required by the TS. The proposed changes to delete non-conservative TS requirements for testing of adsorber carbon is not a plant accident initiator as described in the Final Safety Analysis Report (FSAR). The proposed amendment does not change the function of any structure, system or component (SSC). The function of the ventilation systems is filtration of radiological releases during postulated accidents. The proposed changes will provide greater assurance that this function is provided. The revised TS requirements are for laboratory tests that are currently in place to address Generic Letter 99–02, with one exception to the safety factor of 2, and accommodate the change of the Control Room Ventilation System (CRVS) charcoal beds to two inches. The change only affects the TS testing requirements since the modification to the CRVS will be accomplished separately from the TS change. The TS changes will not result in any changes to the efficiency assumed in accident analysis. The changes do not alter, degrade or prevent actions described or assumed in an accident described in the FSAR. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: The proposed license amendment adopts the new test method and acceptance criteria of ASTM D3803–1989 for activated charcoal filters. The proposed license amendment does not reduce the margin of safety but enhances it by requiring more accurate testing. The proposed test change will require the use of a current and improved ASTM standard to ensure that the carbon ability to adsorb radioactive material will remain at or above the capability credited in our accident analysis.

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

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: January 23, 2003.

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

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

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

The proposed change involves modifications to the TS operability requirements for the ECCS during shutdown conditions. The ECCS is designed to mitigate the release of radioactive materials to the environment following a Loss of Coolant Accident (LOCA). The modifications remove certain ECCS TS requirements during shutdown conditions and includes additional requirements for the Cold Shutdown or Refuel Modes when the availability of the ECCS is most likely to be needed. The additional requirements are more restrictive and are proposed to reduce the probability or consequences of potential accidents. The requirements proposed to be removed are unnecessary due to the associated plant conditions and other changes are administrative in nature. No increase in the probability or consequences of an accident previously evaluated has been identified for these changes. The ECCS is not an initiator of any accidents previously evaluated and the proposed change does not increase the amount of radioactive materials available to be released for a previously evaluated accident. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change involves modifications to the TS operability requirements for the ECCS during shutdown conditions. The modifications remove unnecessary ECCS TS requirements during shutdown conditions and includes additional requirements for the Cold Shutdown or Refuel Modes when the availability of the ECCS is most likely to be needed. In addition, the proposed change makes administrative changes. The proposed change does not involve any physical alteration of ECCS equipment and does not create a new mode of system operation. In addition, no new or different types of ECCS equipment will be installed as a result of the proposed change. The proposed change will allow the installation of modifications on the reference and variable legs of the instrument racks that support the ECCS and Feedwater level instrumentation. No other types of accidents or accident initiators associated with the proposed change or modifications have been identified. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The ECCS is designed to mitigate the release of radioactive materials to the environment following a LOCA. The long-term cooling analysis following a design basis LOCA demonstrates that only one low-pressure ECCS injection/spray subsystem is required, post LOCA, to maintain adequate reactor vessel water level. The proposed change includes an additional requirement that two low-pressure injection/spray subsystems be Operable for the Cold Shutdown or Refuel Modes. The requirements proposed to be removed are unnecessary due to the associated plant conditions and other proposed changes are administrative in nature. No scenario has been identified that, as a result of the proposed change, would create a single component failure which prevents the automatic initiation of the ECCS. The proposed change will not modify the method by which any safety-related system performs its function and ECCS operation and testing will remain consistent with current safety analysis assumptions. 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: December 20, 2002.

Description of amendment request: The proposed amendments would remove technical specification requirements for reactor protection system Function 5, main steam isolation valve closure, and Function 10, turbine condenser vacuum low, when in startup.

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 changes to the Dresden Nuclear Power Station (DNPS) Units 2 and 3 Technical Specifications (TS) revise the applicability of TS 3.3.1.1, ''Reactor Protection System (RPS) Instrumentation,'' Function 5 (*i.e.*, Main Steam Isolation Valve—Closure) and Function 10 (*i.e.*, Turbine Condenser Vacuum—Low) to eliminate the requirement for these functions to be operable while in Mode 2 with reactor pressure ≥600 psig. The proposed changes also delete Required Action F.2 of TS 3.3.1.1 to align with the revised applicability for Functions 5 and 10.

TS requirements that govern operability or routine testing of plant instruments are not assumed to be initiators of any analyzed event because these instruments are intended to prevent, detect, or mitigate accidents. Therefore, these proposed changes will not involve an increase in the probability of an accident previously evaluated.

Additionally, these proposed changes will not increase the consequences of an accident previously evaluated because the proposed changes do not adversely impact structures, systems, or components. These changes will not alter the operation of equipment assumed to be available for the mitigation of accidents or transients by the plant safety analysis. Functions 5 and 10 are currently required in Mode 2 with reactor pressure ≥600 psig to ensure that the reactor is shut down to prevent an overpressurization transient due to closure of main steam isolation valves or turbine stop valves. The existing scram logic is the result of experience gained during the startup of an early vintage boiling water reactor in 1966 when operators had difficulty controlling reactor power above approximately 600 psig without pressure control. Experience on later plant startups indicates that the early experience may not be inherent to the boiling water reactor design. As such, General Electric subsequently recommended that the scram requirement be eliminated. In Mode 2, the heat generation rate is low enough so that the other diverse RPS functions provide sufficient protection from an overpressurization transient. Furthermore, there will be no change in the types or significant increase in the amounts of any effluents released offsite.

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

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 revise the applicability for Functions 5 and 10 of TS 3.3.1.1. The RPS is not an initiator of any accident. Rather, the RPS is designed to initiate a reactor scram when one or more monitored parameters exceed their specified limits to preserve the integrity of the fuel cladding and the reactor coolant pressure boundary and minimize the energy that must be absorbed following an accident. The proposed changes do not alter the applicability for RPS functions during plant conditions in which an overpressurization transient is assumed to occur. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Margins of safety are established in the design of components, the configuration of components to meet certain performance parameters, and in the establishment of setpoints to initiate alarms and actions. The proposed changes revise the applicability for Functions 5 and 10 of TS 3.3.1.1. The proposed changes do not alter the applicability for RPS functions during plant conditions in which an overpressurization transient is assumed to occur. In addition, the proposed changes do not affect the probability of failure or availability of the affected instrumentation. Furthermore, the proposed changes will reduce the probability of test-induced plant transients and equipment failures. 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 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.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit 2, Beaver County, Pennsylvania

Date of amendment request: February 4, 2003.

Description of amendment request: The proposed amendment would extend the surveillance interval of the slave relay in the Engineered Safety Feature Actuation System instrumentation from 92 days to 12 months. The proposed amendment includes changes to surveillance requirement (SR) 4.3.2.1.1 and the related Bases.

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

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

Response: No. The proposed change to the slave relay test interval reduces the potential for spurious actuation of equipment, and therefore does not increase the probability of any accident previously analyzed. The proposed change to the slave relay test interval does not change the response of the unit to any accidents and has an insignificant impact on the reliability of the engineered safety feature actuation system (ESFAS) signals. The ESFAS will remain highly reliable and the proposed change will not result in a significant increase in the risk of plant operation. This is demonstrated by showing that the impact on plant safety as measured by the change in core damage frequency (CDF) is less than 1.0E–06 per year and the change in large early release frequency (LERF) is less than 1.0E–07 per year. The change meets the acceptance criteria in Regulatory Guide 1.174. Therefore, since the ESFAS will continue to perform its function with high reliability as originally assumed, and the increase in risk as measured by the change in CDF and LERF is within the acceptance criteria of existing regulatory guidance, there will not be a significant increase in the consequences of any accidents.

The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the unit is operated and maintained. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed change does not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed change is consistent with the safety analysis assumptions and resultant consequences.

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

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

Response: No. The proposed change does not result in a change in the manner in which the EFSAS provides unit protection. The EFSAS will continue to have the same setpoints after the proposed change is implemented. There are no design changes associated with the proposed change. The change to the slave relay test interval does not change any existing accident scenarios, nor create any new or different accident scenarios.

The change does not involve a physical alteration to the unit (*i.e.*, no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the change does not impose any new or different requirements or eliminate any existing requirements. The change does not alter assumptions made in the safety analysis. The proposed change is consistent with the safety analysis assumptions and current unit operating practice.

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

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

Response: No. The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not impacted by this change. Redundant ESFAS trains are maintained, and diversity with regard to the signals that provide engineered safety features actuation is also maintained. All signals credited as primary or secondary, and all operator actions credited in the accident analysis will remain the same. The proposed change will not result in unit operation in a configuration outside the design basis. The calculated impact on risk is insignificant and meets the acceptance criteria contained in Regulatory Guide 1.174. The proposed slave relay test interval change will result in a reduced potential for spurious equipment actuations associated with testing.

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

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

Attorney for licensee: Mary O'Reilly, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: June 10, 2002.

Description of amendment request: The proposed amendment would revise Surveillance Requirement (SR) 3.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period would be extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement would be added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, 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 September 28, 2001 (66 FR 49714). The licensee affirmed the applicability of the following NSHC determination in its application dated June 10, 2002.

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

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

The proposed change relaxes the time allowed to perform a missed surveillance. The time between surveillances is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Any reduction in confidence that a standby system might fail to perform its safety function due to a missed surveillance is small and would not, in the absence of other unrelated failures, lead to an increase in consequences beyond those estimated by existing analyses. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. A missed surveillance will not, in and of itself, introduce new failure modes or effects and any increased chance that a standby system might fail to perform its safety function due to a missed surveillance would not, in the absence of other unrelated failures, lead to an accident beyond those previously evaluated. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The extended time allowed to perform a missed surveillance does not result in a significant reduction in the margin of safety. As supported by the historical data, the likely outcome of any surveillance is verification that the LCO [Limiting Condition for Operation] is met. Failure to perform a surveillance within the prescribed frequency does not cause equipment to become inoperable. The only effect of the additional time allowed to perform a missed surveillance on the margin of safety is the extension of the time until inoperable equipment is discovered to be inoperable by the missed surveillance. However, given the rare occurrence of inoperable equipment, and the rare occurrence of a missed surveillance, a missed surveillance on inoperable equipment would be very unlikely. This must be balanced against the real risk of manipulating the plant equipment or condition to perform the missed surveillance. In addition, parallel trains and alternate equipment are typically available to perform the safety function of the equipment not tested. Thus, there is confidence that the equipment can perform its assumed safety function.

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

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

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: February 28, 2003.

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) to relocate the numerical values and curves for the pressure and temperature (P/T) limits for the reactor coolant system (RCS). The numerical values and curves would be relocated from the TS to a licensee-controlled document, the Pressure and Temperature Limits Report (PTLR) pursuant to Nuclear Regulatory Commission (NRC) Generic Letter (GL) 96–03, ''Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits,'' dated January 31, 1996, as modified by NRC Improved Standard TS, TS Task Force (TSTF) change package number 419, Revision 0. Specifically, a definition for the PTLR would be added to TS 1.0, ''Definitions;'' administrative controls for the generation and reporting requirements associated with the PTLR would be added to TS 5.6, ''Administrative Controls—Reporting Requirements; '' TSs 3.4.9 and 4.4.9 would be modified by removing the numerical values and curve (Figure 3.4.9–1) for the various P/T limits (which the licensee has updated using an NRC-approved methodology) and replacing them with a reference to the PTLR.

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

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed by the ASME [American Society of Mechanical Engineers Boiler and Pressure Vessel] Code and 10 CFR [Part] 50 Appendi[ces] G and H as restrictions on normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause non-ductile failure of the reactor coolant pressure boundary. Thus, they ensure that an accident precursor is not likely. Hence, they are included in the TS as satisfying Criterion 2 of 10 CFR 50.36(c)(2)(ii). The relocation of the numerical value of these limits to a licensee-controlled document does not remove the existing TS requirement that the limits be met. The new TS administrative controls for the PTLR will ensure that only NRC-approved methods are used to calculate the actual limits to be applied. Thus, this relocation will not increase the probability of any accident previously evaluated.

The proposed changes do not alter the design assumptions, conditions, or configuration of the facility or the manner in which the facility is operated or maintained. The proposed changes will not affect any other System, Structure or Component (SSC) designed for the mitigation of previously analyzed events. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of any accident previously evaluated. Thus, the proposed relocation of the existing numerical values and the updated figure for the RCS P/T limits based upon an NRC-approved methodology, to a licensee-controlled document (*i.e.*, the PTLR), with all the requisite TS restrictions placed upon it by NRC Generic Letter 96–03, as modified by TSTF–419, Rev. 0, will not increase the consequences of any previously evaluated accident.

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

The proposed changes do not involve a physical alteration of the plant (*i.e.*, no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice. We are merely requesting to move the existing numerical values and the updated figure for the RCS P/T limits based upon an NRC-approved methodology, from the TS to a licensee-controlled document (*i.e.*, the PTLR), with all the requisite TS restrictions placed upon it by NRC Generic Letter 96–03, as modified by TSTF–419, Rev. 0.

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

(3) The proposed amendment will not involve a significant reduction in a margin of safety.

The proposed changes do not alter the manner in which Safety Limits, Limiting Safety System Settings or Limiting Conditions for Operation are determined. The setpoints at which protective actions are initiated are not altered by the proposed changes. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. We are merely requesting to move the existing numerical values and the updated figure for the RCS P/T limits based upon an NRC-approved methodology, from the TS to a licensee-controlled document (*i.e.*, the PTLR), with all the requisite TS restrictions placed upon it by NRC Generic Letter 96–03, as modified by TSTF–419, Rev. 0. Thus, the proposed changes will not significantly reduce any margin of safety that currently exists.

Based upon the above, NMC [Nuclear Management Company] has determined that the proposed amendment will not involve a significant hazards consideration.

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

Attorney for licensee: Mr. Alvin Gutterman, Morgan Lewis, 1111 Pennsylvania Avenue NW Washington, DC 20004.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: January 27, 2003.

Description of amendment request: The proposed amendment would make administrative and editorial changes to the Fort Calhoun Station (FCS) Technical Specifications (TS) 1.3 Basis (1); 2.7(1)a; 2.7(1)b; 2.7(1)d; 2.7(1)i; 2.7 Basis; 3.0.2; Table 3–5, Item 11; and 3.5(3)ii. The proposed changes consist primarily of editorial and typographical changes or corrections.

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

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

The correction of typographical errors and clarification of specifications is not an initiator of any previously evaluated accident. The frequency or periodicity of performance of those surveillances affected by this change are not an initiator of any previously evaluated accident. The proposed changes will not prevent safety systems from performing their accident mitigation function as assumed in the safety analysis.

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

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

The proposed change only affects the technical specifications and does not involve a physical change to the plant. Modifications will not be made to existing components nor will any new or different types of equipment be installed. The proposed change corrects typographical errors, provides clarification as to applicable equipment and modifies the frequency of surveillances performed once per shift from 8 hours to 12 hours. This change will not alter assumptions made in safety analysis and licensing bases.

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

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

The proposed change corrects typographical errors, provides clarification as to applicable equipment, and modifies the frequency of surveillances performed once per shift from 8 hours to 12 hours. The decrease in frequency or periodicity of performance of these surveillances will also permit more efficient and more safely managed plant operations and can help reduce the risk associated with changing plant equipment or operating modes in order to obtain some of these readings.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: January 27, 2003.

Description of amendment request: The proposed amendment would delete the allowance to perform the surveillance test of Table 3–2, Item 20 (Recirculation Actuation Logic Channel Functional Test) under administrative controls, while components in excess of those allowed by Conditions a, b, d, and e of Technical Specification 2.3(2) are inoperable provided they are returned to operable status within one hour. This allowance was granted in Amendment No. 206 issued April 19, 2002, on an exigent basis and applies only for the remainder of the current cycle. Omaha

Public Power District committed to submit a permanent resolution to this allowance and this license amendment request constitutes this permanent resolution.

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

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

Deleting the requirement to perform the quarterly surveillance test of Table 3–2, Item 20 (Recirculation Actuation Logic Channel Functional Test) under administrative controls is acceptable since the performance of the recirculation actuation logic channel functional test is not identified as the initiator of any analyzed event. The proposed change will still require that the surveillance test be performed and the required ECCS [emergency core cooling system] systems to be available. This change will not alter assumptions relative to the mitigation of an accident or transient event. The performance of this activity has no effect on any accident scenario. Therefore, the proposed change does not involve a significant increase in the 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.

This change only removes a short term allowance to utilize administrative controls in the performance of the recirculation actuation logic channel functional test. These proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or change the methods governing plant operation. The proposed change does not involve any physical changes to plant systems, structures or components (SSCs) or the manner in which these SSCs are operated, maintained, modified or inspected. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The minimum numbers of ECCS components required by the FCS [Fort Calhoun Station] accident analyses will remain available. The proposed change to delete the short term allowance to utilize administrative controls in the performance of the recirculation actuation logic channel functional test will not significantly impact the availability or reliability of the plant's systems or their ability to respond to plant transients and accidents. The performance of this activity has no effect on any accident scenario. 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: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: January 27, 2003.

Description of amendment request: The proposed amendment would authorize the revision of the Fort Calhoun Station, Unit No. 1 Updated Safety Analysis Report (USAR). Section 14.16 and Figures 14.16–1 through 14.16–4 of the USAR will be revised to reflect the use of the GOTHIC, version 7.0, computer code and the results associated with the updated containment pressure analyses for a loss-of-coolant accident and main steam line break. In addition, GOTHIC will be used for the analysis of future plant upgrades associated with containment response and will be maintained consistent with other NRC-approved Omaha Public Power District methodologies.

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

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

The proposed changes will not increase the probability or consequence of any accident based on the following:

The proposed changes to Section 14.16 of the Updated Safety Analysis Report (USAR) and replacements for Figures 14.16–1 through 14.16–4 is required due to using GOTHIC, version 7.0 and the updated containment pressure analyses. Demonstrating that containment pressure is maintained less than the containment design pressure is required by Fort Calhoun Station (FCS) design basis. Additionally, the analyses credit all modes of heat transfer defined by Reference 10.5. Therefore, the updated containment pressure analyses using GOTHIC, version 7.0 is in compliance with FCS design basis. Changes to the containment pressure analyses for either a loss-of-coolant accident or main steam line break will be controlled by 10 CFR 50.59. Therefore, the probability or consequence of any accident is not increased.

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

The proposed revision does not change any equipment required to mitigate the consequences of an accident. The continued use of the same USAR administrative controls prevents the possibility of a new or different kind of accident. Since the proposed changes do not involve the addition or modification of equipment nor alter the design of plant systems, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes proposed do not change how design basis accident events are postulated nor do the changes themselves initiate a new kind of accident or failure mode with a unique set of conditions (proposed administrative controls). Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The use of GOTHIC, version 7.0 is in compliance with FCS design basis. Additionally, GOTHIC has been benchmarked to the current analysis of record for a loss-of-coolant accident and main steam line break using the NRC approved computer code CONTRANS. These benchmark models demonstrate that GOTHIC provides similar results to CONTRANS. Future updates of the containment pressure analyses will be conducted under the 10 CFR 50.59 process. The analyses will credit all available modes of heat transfer defined by Reference 10.5. Additionally, the main steam line break containment evaluation model considers the leakage past the broken steam generator main feed isolation valve of 2.45% of full power flow or approximately 195 gpm. Therefore, the proposed changes do not involve a significant reduction to the margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: January 27, 2003.

Description of amendment request: The proposed amendment revises Technical Specifications (TS) 2.1.6, 3.2

(Table 3–5), and 5.9.1c. For TS 2.1.6(1), Omaha Public Power District (OPPD) has proposed to increase the ''as-found'' pressurizer safety valve (PSV) lift setting tolerance band of ±1% to +1%/–3% to allow for normal setpoint variance for Modes 1 and 2. The Basis of TS 2.1.6 will be revised to clarify that the PSVs are still operable and capable of performing their safety function with the wider tolerance band. The remaining revisions to TS 2.1.6 are administrative in nature to change defined terms to upper case text. OPPD has also proposed to revise (1) item 3 in Table 3–5 of TS 3.2 to require an ''as-left'' PSV lift setting tolerance band of ±1%, and (2) TS 5.9.1c to remove the requirement to provide a statement in the Monthly Operating Report (MOR) concerning failures or challenges to power operated relief valves (PORV) or safety valves.

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

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

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

The design basis event for RCS over-pressure protection is the Loss of Load accident. The Loss of Load event was previously evaluated assuming the PSVs lift up to 6% above their setpoint. While the proposed amendment widens the tolerance band for installed PSVs, only the lower end of the band is changed; therefore, there is no adverse affect on the over-pressure protection analysis.

The proposed amendment does not change the tolerance band currently required at the conclusion of PSV surveillance testing each refueling outage. As with the current specification, the PSVs will continue to be set to within a tolerance band of ± 1% using ASME Code test methods. As a result, the anticipated performance of the valves over the course of the subsequent operating cycle is not changed. In other words, the potential for setpoint variance exists regardless of whether the TSs are changed. The PSVs will begin each operating cycle after having been set to open within a lift setting tolerance band of ± 1%. Therefore, the probability or consequences of potential setpoint variance during an operating cycle does not change. The remaining changes provide supporting statements for the wider PSV lift setting tolerance band in the Basis of TS 2.1.6, are administrative in nature, or are in accordance with GL 97–02.

The changes in the case of the defined terms and elimination of the TS 5.9.1c Monthly Operating Report concerning failures or challenges to PORVs or safety valves are administrative changes which do not affect the initiator of an event or prevent safety systems from performing their accident mitigation functions as assumed in the safety analysis.

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

Widening the lift setting tolerance band for installed PSVs does not create the possibility of a new or different type of accident from any previously evaluated.

The accident analyses address the lift setting tolerance band of the PSVs, and the proposed tolerance band does not adversely affect the over-pressure protection function and will not compromise RCS integrity during power operation. No physical changes to the plant are involved.

The proposed amendment does not change the tolerance band that must be met at the conclusion of PSV surveillance testing each refueling outage. As with the current Technical Specifications, the PSVs will continue to be set at a tolerance band of ± 1% using ASME Code test methods. As a result, the anticipated performance of the valves over the course of the subsequent operating cycle is not changed. The remaining changes provide supporting statements for the wider PSV lift setting tolerance band in the Basis of TS 2.1.6, are administrative in nature, or are in accordance with GL 97–02 and thus do not create the possibility of a new or different type of accident from any previously evaluated.

The changes in the case of the defined terms and elimination of the TS 5.9.1c Monthly Operating Report concerning failures or challenges to PORVs or safety valves are administrative changes which only affect the technical specifications and do not involve a physical change to the plant. Therefore these changes do not alter assumptions made in the safety analysis and licensing basis.

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

Widening the lift setting tolerance band for installed PSVs does not involve a significant reduction in a margin of safety. The tolerance band of the PSVs is addressed in the accident analyses, and the proposed tolerance band does not adversely affect the over-pressure protection analysis. No physical changes to the plant are involved.

The proposed amendment does not change the tolerance band that must be met at the conclusion of PSV surveillance testing each refueling outage. As with the current Technical Specifications, the PSVs will continue to be set to a tolerance band of ± 1% using ASME Code test methods. As a result, the anticipated performance of the valves over the course of the subsequent operating cycle is not changed. The remaining changes provide supporting statements for the wider PSV lift setting tolerance band in the Basis of TS 2.1.6, are administrative in nature, or are in accordance with GL 97–02.

The changes in the case of the defined terms and elimination of the TS 5.9.1c Monthly Operating Report concerning failures or challenges to PORVs or safety valves are administrative changes which only affect the technical specifications and reporting frequency.

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

NRC Section Chief: Stephen Dembek.

Tennessee Valley Authority, Docket Nos. 50–259, 50–260 and 50–296, Browns Ferry Nuclear Plant, Units 1, 2 and 3, Limestone County, Alabama

Date of amendment request: February 19, 2003.

Description of amendment request: The proposed amendments delete 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. 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 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 February 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 requests involve no significant hazards consideration.

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

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: December 13, 2003.

Description of amendment request: The proposed amendment would allow the use of Westinghouse leak-limiting Alloy 800 sleeves to repair defective steam generator tubes as an alternative to plugging the tube.

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 in accordance with the three standards set forth in 10 CFR 50.92(c), which are presented below:

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

No. The Westinghouse Alloy 800 leak-limiting repair sleeves are designed using the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and, therefore, meet the design objectives of the original steam generator tubing. The applied stresses and fatigue usage for the repair sleeves are bounded by the limits established in the ASME Code. Mechanical testing has shown that the structural strength of repair sleeves under normal, upset, emergency, and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (three times normal operating pressure differential) burst margin recommended by NRC's Regulatory Guide 1.121, ''Bases for Plugging Degraded PWR Steam Generator Tubes.'' Burst testing of sleeve/tube assemblies has demonstrated that no unacceptable levels of primary-to-secondary leakage are expected during any plant condition.

The Alloy 800 repair sleeve depth-based structural limit is determined using the NRC guidance and the pressure stress equation of ASME Code, Section III with additional margin added to account for configuration of long axial cracks. A bounding detection threshold value has been conservatively identified and statistically established to account for growth and determine the repair sleeve/tube assembly plugging limit. A sleeved tube is plugged on detection of degradation in the sleeve/tube assembly.

Evaluation of the repaired steam generator tube testing and analysis indicates no detrimental effects on the sleeve or sleeved tube assembly from reactor system flow, primary or secondary coolant chemistries, thermal conditions or transients, or pressure conditions as may be experienced at Watts Bar Unit 1. Corrosion testing and historical performance of sleeve/tube assemblies indicates no evidence of sleeve or tube corrosion considered detrimental under anticipated service conditions.

The implementation of the proposed amendment has no significant effect on either the configuration of the plant or the manner in which it is operated. The consequences of a hypothetical failure of the sleeve/tube assembly is bounded by the current steam generator tube rupture (SGTR) analysis described in Watts Bar Unit 1 Updated Final Safety Analysis Report. Due to the slight reduction in diameter caused by the sleeve wall thickness, primary coolant release rates would be slightly less than assumed for the steam generator tube rupture analysis and; therefore, would result in lower total primary fluid mass release to the secondary system. A main steam line break or feedwater line break will not cause a SGTR since the sleeves are analyzed for a maximum accident differential pressure greater that that predicted in the Watts Bar Unit 1 safety analysis. The minimal repair sleeve/tube assembly leakage that could occur during plant operation is well within the Technical Specification leakage limits when grouped with current alternate plugging criteria calculated leakage values.

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

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

No. The Alloy 800 leak-limiting repair sleeves are designed using the applicable ASME Code as guidance; therefore, it meets the objectives of the original steam generator tubing. As a result, the functions of the steam generators will not be significantly affected by the installation of the proposed sleeve. The proposed repair sleeves do not interact with any other plant systems. Any accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing SGTR accident analysis. The continued integrity of the installed sleeve/tube assembly is periodically verified by the Technical Specification requirements and the sleeved tube plugged on detection of degradation.

The implementation of the proposed amendment has no significant effect on either the configuration of the plant, or the manner in which it is operated. Therefore, TVA concludes that this 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?

No. The repair of degraded steam generator tubes with Alloy 800 leak-limiting repair sleeves restores the structural integrity of the degraded tube under normal operating and postulated accident conditions and thereby maintains current core cooling margin as opposed to plugging the tube and taking it out of service. The design safety factors utilized for the repair sleeves are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in the original steam generator design. The portions of the installed sleeve/tube assembly that represent the reactor coolant pressure boundary can be monitored for the initiation of sleeve/tube wall degradation and affected tube plugged on detection. Use of the previously identified design criteria and design verification testing assures that the margin to safety is not significantly different from the original steam generator tubes.

Therefore, TVA concludes that 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902.

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: December 13, 2002.

Description of amendment request: The proposed amendment would revise the Watts Bar Nuclear Plant, Unit 1, Technical Specifications to add two new Sections, 3.7.16, ''Shutdown Board Room Air Conditioning System,'' and 3.7.17, ''Elevation 772.0 480 Volt Board Room Air Conditioning Systems.''

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 in accordance with the three standards set forth in 10 CFR 50.92(c), which are presented below:

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

[No.] The proposed revision to the [Watts Bar Nuclear Plant] TS will provide formalized operational guidance for coping with partial or complete unavailability of SDBR [shutdown board room] and 480V board room air conditioning (AC) equipment for limited periods of time. The change does not impact the frequency of an accident because failure of either the SDBR or the 480V board room AC systems is not an initiator of any accident scenario. The change does not modify any plant hardware including the air conditioning systems, and none of their automatic control features or redundant systems currently credited in failure analyses are being deleted, modified, or otherwise replaced by operator actions as a result of the proposed change.

The proposed TS revision changes current plant operating practice and WBN Final Safety Analysis Report (FSAR) assumptions by allowing continued power operation with both trains of SDBR air conditioning concurrently inoperable and two 480V board room AC systems of the same unit to be concurrently inoperable for a limited duration, up to 12 hours. This condition is acceptable based on the low probability of the occurrence of postulated accidents resulting in core damage concurrent with multiple inoperable systems or trains of cooling equipment during this timeframe, and based on analyses which demonstrate that peak temperatures in each room served by these systems remain below mild environment temperature limits during this time period. Consequently, there is no significant adverse impact on the ability of required safety-related electrical equipment to continue to operate and perform their required functions, during both normal operation and during design basis events. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

[No.] The proposed change does not modify any plant hardware including the subject air conditioning systems. The change provides specific operational guidance for coping with partial or complete unavailability of SDBR and 480V board room air conditioning equipment. No new accident or event initiators are created by allowing multiple air conditioning systems to be unavailable for the limited time period of 12 hours. The supported electrical equipment remains capable of performing its intended function both during normal operations and post accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

[No.] The proposed TS revision changes current FSAR assumptions by allowing continued power operation with both trains of SDBR air conditioning concurrently inoperable and allowing two 480V board room air conditioning systems of the same unit to be inoperable for a limited duration, up to 12 hours. This condition does not significantly reduce the margin of safety due to the low probability of the occurrence of a postulated accident resulting in core damage concurrent with multiple inoperable systems or trains of cooling equipment during the limited time period. In addition, transient temperature analyses demonstrate that peak temperatures in each room served by these systems remain below mild environment temperature limits for a period of 24 hours assuming a complete loss of air conditioning to all rooms served by the SDBR and 480V board room AC systems concurrently. The analysis is bounding for normal operational conditions. Consequently, there is no significant adverse impact on the ability of required safety-related electrical equipment to continue to operate and perform their required functions during both normal operation and during design basis events. 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902.

NRC Section Chief: Allen G. Howe.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action *see* (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, 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.*

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

Date of application for amendments: September 20, 2002.

Brief description of amendments: These amendments adopt the generic changes approved by Technical Specification Task Force (TSTF) change travelers TSTF–349, Revision 1, and TSTF–361, Revision 2, for NUREG–1430, Revision 1, ''Standard Technical Specifications, Babcock and Wilcox Plants,'' dated April 1995, and incorporated into NUREG–1430, Revision 2, dated June 2001. Specifically, Section 3.9.5, ''Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level,'' is revised to add two notes to allow operational changes in the shutdown cooling system.

Date of issuance: February 25, 2003.

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

Amendment Nos.: 256 and 233.

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

*Date of initial notice in **Federal Register:*** October 29, 2002 (67 FR 66007).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 5, 2002, as supplemented January, 14, 2003.

Brief description of amendment: The amendment revises the surveillance requirements associated with the Containment Isolation Valves (CIVs), Reactor Building Closed Cooling Water (RBCCW) System, and Service Water (SW) System to remove redundant testing requirements that are already addressed by the Inservice Testing Program. Additional changes remove the post maintenance testing requirements associated with the CIVs, revise the wording of the RBCCW and SW Systems Limiting Conditions for Operation, and increase the allowed outage times for the RBCCW and SW Systems.

Date of issuance: February 13, 2003.

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

Amendment No.: 273.

Facility Operating License No. DPR–65: This amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** April 16, 2002 (67 FR 18644). The January 14, 2003, letter 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 February 13, 2003.

No significant hazards consideration comments received: No.

Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: May 14, 2002, as supplemented by letter dated December 20, 2002.

Brief description of amendment: The amendment changes administrative Technical Specification 5.5.13 regarding the Containment Integrated Leak Rate Testing (ILRT) to allow a one-time extension of the interval (to 15 years) for performance of the next ILRT.

Date of issuance: March 5, 2003.

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

Amendment No.: 131.

Facility Operating License No. NPF–47: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** June 25, 2002 (67 FR 42823). The December 20, 2002, supplemental letter provided clarifying information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 5, 2003.

No significant hazards consideration comments received: No.

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of application for amendment: December 10, 2002, as supplemented on January 20, 2003.

Brief description of amendment: The Technical Specification (TS) amendment request changes the diesel fuel specification to a more current revision in TS 4.10.C. The changes also make administrative revisions to reflect generic position titles in TS 6.0; correct page numbers and titles in the Table of Contents; and to delete the General Table of Contents. Bases pages were also revised to reflect the fuel specification revision, as well as to make administrative changes to provide clarity and correct a misspelling.

Date of Issuance: February 27, 2003.

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

Amendment No.: 214.

Facility Operating License No. DPR–28: Amendment revised the Technical Specifications.

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

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated February 27, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 27, 2002.

Brief description of amendments: The amendments change Appendix B, ''Environmental Protection Plan,'' of the licensee by removing a parenthetical reference to a superseded section of 10 CFR part 51.

Date of issuance: February 20, 2003.

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

Amendment Nos.: 157/143

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Environmental Protection Plan.

Date of initial notice in ***Federal Register:*** October 29, 2002 (67 FR 66009).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 20, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 7, 2002.

Brief description of amendments: The amendments: (1) Revised the surveillance frequency for air or smoke flow testing of containment spray nozzles, as specified in surveillance requirements (SRs) 4.6.2.1.d and 4.6.2.2.f, from, ''once per 10 years,'' to, ''following maintenance which results in the potential for nozzle blockage as determined by engineering evaluation;'' (2) allowed the use of a visual examination in lieu of an air or smoke flow test; (3) relocated the SR 4.6.2.2.e.3 criteria for the river/service water flow rate through the recirculation spray system heat exchangers to the Updated Final Safety Analysis Report; and (4) made minor clarifying changes to the text in TS 3.3.1.1.

Date of issuance: February 24, 2003.

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

Amendment Nos.: 252 and 132.

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

Date of initial notice in ***Federal Register:*** October 15, 2002 (67 FR 63694).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 14, 2002.

Brief description of amendments: The amendments revised the Technical Specifications (TSs) by extending the allowed outage time (AOT), or completion time, associated with an inoperable emergency core cooling system (ECCS) accumulator. In addition to the AOT extension, other changes were incorporated to make the ECCS TSs consistent with NUREG–1431, ''Standard Technical Specifications—Westinghouse Plants.'' Format and editorial changes were included as necessary to facilitate the revision of the TS text to conform to the current TS page format.

Date of issuance: February 25, 2003.

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

Amendment Nos.: 253 and 133.

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

Date of initial notice in ***Federal Register:*** April 30, 2002 (67 FR 21289).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 31, 2002, as supplemented by letters dated December 2, 2002, and January 24, 2003.

Brief description of amendments: The amendments revised the Technical Specifications to allow extending the Type A containment integrated leak rate test interval from 10 years to 15 years on a one-time basis.

Date of issuance: March 5, 2003.

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

Amendment Nos.: 254 and 134.

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

Date of initial notice in ***Federal Register:*** December 10, 2002 (67 FR 75877).

The December 2, 2002, and January 24, 2003, supplemental letters did not change the initial no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 5, 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: December 9, 2002.

Brief description of amendment: Pursuant to 10 CFR 50.67, this amendment approves the use of Alternative Source Term radiological calculations to update the design bases analysis for the Fuel Handling Accident as described in the Updated Safety Analysis Report. Regulatory Guide 1.183, ''Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors,'' was used in the application.

Date of issuance: March 4, 2003.

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

Amendment No.: 122.

Facility Operating License No. NPF–58: This amendment revised the Updated Safety Analysis Report.

Date of initial notice in ***Federal Register:*** January 7, 2003 (68 FR 804).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 16, 2002.

Brief description of amendments: The proposed amendments modified Technical Specification (TS) Surveillance Requirement Section 4.0.3 to extend the delay time for completion of a missed surveillance to 24 hours or up to the surveillance frequency, whichever is greater. Additionally the proposed change would add a TS Bases Control Program.

Date of issuance: March 3, 2003.

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

Amendment Nos: 222 and 217.

Facility Operating License Nos. DPR–31 and DPR–41: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 24, 2002 (67 FR 78521).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 3, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 21, 2002, as supplemented by letters dated February 11, 2003, and March 3, 2003.

Brief description of amendments: The amendments will reduce the minimum time required for reactor subcriticality prior to removing irradiated fuel from the reactor vessel from 100 hours to 72 hours, as specified in Technical Specification 3/4.9.3 ''Refueling Operations, Decay Time.''

Date of issuance: March 4, 2003.

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

Amendment Nos: 223 and 218.

Facility Operating License Nos. DPR–31 and DPR–41: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** November 12, 2002 (67 FR 68738).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 4, 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: April 11, 2002, as supplemented November 11, 2002.

Brief description of amendments: The amendments would revise the Surveillance Requirements for containment leakage rate testing in Technical Specification 4.6.1.2 to allow a one-time extension of the interval between integrated leakage rate tests from 10 to 15 years.

Date of issuance: February 25, 2003.

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

Amendment Nos.: 274 and 254.

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

*Date of initial notice in **Federal Register:*** May 14, 2002 (67 FR 34488).

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

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

No significant hazards consideration comments received: No.

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

Date of amendment request: February 28, 2001, as supplemented by letters dated February 26, September 13 and 27, and November 25, 2002 (2).

Brief description of amendment: The amendment consists of changes to the design-basis accidents dose assessment methodology and Operating License Condition 2.C.(6).

Date of issuance: February 21, 2003.

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

Amendment No.: 196.

Facility Operating License No. DPR–46: Amendment revised the final safety analysis report and Operating License Condition 2.C.(6).

*Date of initial notice in **Federal Register:*** September 19, 2001 (66 FR 48289).

The supplemental letters provided clarifying information that was within the scope of the original **Federal Register** notice (66 FR 48289) and did not change the initial no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 3, 2003.

Brief description of amendment: The amendment changed Technical Specifications Surveillance Requirement 3.6.1.7.2 for suppression chamber-to-drywell vacuum breaker 2ISC*RV36B to allow an exception to the periodic functional testing requirements for the remainder of Cycle 9.

Date of issuance: February 21, 2003.

Effective date: As of the date of issuance to be implemented within 7 days.

Amendment No.: 108.

Facility Operating License No. NPF–69: Amendment revises the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: Yes. The Nuclear Regulatory Commission published a public notice of the proposed amendment, issued a proposed finding of no significant hazards consideration and requested that any comments on the proposed no significant hazards consideration be provided to the staff by the close of business on February 20, 2003. The notice was published in the Syracuse, NY, *The Post-Standard,* on February 11, 2003.

No significant hazards consideration comments received: No.

The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of New York, and final no significant hazards consideration determination are contained in a Safety Evaluation dated February 21, 2003.

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

Date of application for amendment: April 22, 2002, as supplemented September 16, 2002.

Brief description of amendment: The amendment changes the Technical Specifications by revising the curves for minimum pressure-temperature for the reactor pressure vessel. The P–T curves addressed by this amendment were developed in accordance with (1) the 1989 edition of the American Society of Mechanical Engineers (ASME) Code, section Xl, appendix G, (2) 10 CFR part 50, appendix G, and (3) ASME Code Case N–640, ''Alternative Reference Fracture Toughness for Development of P–T Limit Curves.''

Date of issuance: February 24, 2003.

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

Amendment No.: 133.

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

*Date of initial notice in **Federal Register:*** September 3, 2002 (67 FR 56323).

The September 16, 2002, supplemental letter provided additional clarifying information that was within the scope of the original application, did not change the NRC staff's 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 February 24, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 1, 2002, as supplemented November 7, 2002.

Brief description of amendment: The amendment revises the testing frequency for the containment spray nozzles specified in Technical Specification Surveillance Requirement 3.6.6.9. The testing frequency for the containment spray nozzles is changed from 10 years to ''following maintenance which could result in nozzle blockage.''

Date of issuance: February 24, 2003.

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

Amendment No.: 211.

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

*Date of initial notice in **Federal Register:*** October 15, 2002 (67 FR 63696).

The November 7, 2002, supplemental letter provided additional clarifying information that was within the scope of the original application, did not change the NRC staff's 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 February 24, 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.

Brief description of amendment: The amendment relocates the requirements of TS 3.5(5) for testing prestressed concrete containment tendons to the Fort Calhoun Station, Unit No. 1 Updated Safety Analysis Report. The amendment adds the requirement for a Containment Tendon Testing Program (TS 5.21) consistent with that presented in Section 5.5 of NUREG–1432, ''Improved Standard Technical Specification (ITS) for Combustion Engineering Plants.''

Date of issuance: February 26, 2003.

Effective date: February 26, 2003, and shall be implemented within 120 days from the date of issuance, including the incorporation of the containment tendons testing requirements into the Updated Safety Analysis Report.

Amendment No.: 216.

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

*Date of initial notice in **Federal Register:*** November 12, 2002 (67 FR 68741).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 26, 2003.

No significant hazards consideration comments received: No.

PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: October 16, 2001, as supplemented August 23, 2002, November 8, 2002, and January 20, 2003.

Brief description of amendments: These amendments revised the technical specifications (TSs) to incorporate seven industry-proposed Technical Specification Task Force changes (TSTFs) made to NUREG–1433, Revision 1, ''Standard Technical Specifications for General Electric Plants (BWR/4),'' that have been approved by the Nuclear Regulatory Commission.

Date of issuance: February 25, 2003.

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

Amendment Nos.: 209 and 183.

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

*Date of initial notice in **Federal Register:*** December 12, 2001 (66 FR 64300). The supplements dated August 23, 2002, November 8, 2002, and January 20, 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 February 25, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 17, 2002, as supplemented by letters dated October 30, 2002, December 18, 2002, and January 28, 2003.

Brief description of amendments: The amendment revised the values of the Safety Limit for Minimum Critical Power Ratio in the Unit 2 Technical Specifications (TSs) 2.1.1.2, clarified fuel design features in TS 4.2.1, and updated the references used to determine the core operating limits in TS 5.6.5.b.

Date of issuance: March 4, 2003.

Effective date: As of the date of issuance and shall be implemented upon startup following the Susquehanna Steam Electric Station, Unit 2 eleventh refueling and inspection outage.

Amendment Nos.: 184.

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

*Date of initial notice in **Federal Register:*** August 20, 2002 (67 FR 53988).

The supplements dated October 30, 2002, December 18, 2002, and January 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 staff's original proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 2, 2002.

Brief description of amendments: The amendments revised Technical Specification Surveillance Requirement 3.6.4.1.2 to require that only one access door in each opening of the secondary containment be closed.

Date of issuance: February 28, 2003.

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

Amendment Nos.: 236/178.

Renewed Facility Operating License Nos. DPR–57 and NPF–5: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 7, 2003 (68 FR 812).

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

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–260, Browns Ferry Nuclear Plant, Unit 2, Limestone County, Alabama

Date of application for amendments: October 25, 2002, as supplemented December 20, 2002, and February 11 and 21, 2003.

Description of amendment request: The amendment updated the values of the Safety Limit Minimum Critical Power Ratio in Technical Specification 2.1.1.2 for Cycle 13 operation.

Date of issuance: February 28, 2003.

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

Amendment No.: 280.

Facility Operating License No. DPR–52: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 10, 2002 (67 FR 75885). The supplemental letters 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 amendment is contained in a Safety Evaluation dated February 28, 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: March 29, 2002, as supplemented on October 10, 2002.

Brief description of amendment: The proposed amendment deletes several of the Unit 1 Technical Specification (TS) Surveillance Requirements (SR) contained in TS 3/4.4.5, ''Steam Generators'' (SGs), associated with the voltage-based SG alternative repair criteria. In addition the proposed changes would delete License Condition 2.C.9.d which references commitment letters associated with SG inspection activities.

Date of issuance: March 4, 2003.

Effective date: As of the date of issuance and shall be implemented during the 2003 Cycle 12 Refueling Outage.

Amendment No.: 282.

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

Date of initial notice in ***Federal Register:*** August 6, 2002 (67 FR 50960). An October 10, 2002 submittal revised some of the information, so a revised notice was published October 29, 2002 (67 FR 66014).

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

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 10th day of March, 2003.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 03–6286 Filed 3–17–03; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

[Investment Company Act Release No. 25956; 812–12274]

JNL Series Trust, *et al.*; Notice of Application

March 12, 2003.

AGENCY: Securities and Exchange Commission (''Commission'').

ACTION: Notice of an application under section 6(c) of the Investment Company Act of 1940 (the ''Act'') for an exemption from section 15(a) of the Act and rule 18f-2 under the Act, as well as from certain disclosure requirements.

SUMMARY OF APPLICATION: The requested order would permit certain registered open-end management investment companies to enter into and materially amend subadvisory agreements without shareholder approval and grant relief from certain disclosure requirements.

APPLICANTS: Jackson National Asset Management, LLC (the ''Manager''), JNL Series Trust (''Series Trust''), JNL Investors Series Trust (''Investors Series Trust''), and JNL Variable Fund LLC, JNL Variable Fund III LLC, JNL Variable Fund V LLC, JNLNY Variable Fund I LLC and JNLNY Variable Fund II LLC (collectively, the ''Variable Funds'').

FILING DATES: The application was filed on September 22, 2000 and amended on December 27, 2001 and March 6, 2003.

HEARING OR NOTIFICATION OF HEARING: An order granting the application will be issued unless the Commission orders a hearing. Interested persons may request a hearing by writing to the Commission's Secretary and serving applicants with a copy of the request, personally or by mail. Hearing requests should be received by the Commission by 5:30 p.m. on April 7, 2003, and should be accompanied by proof of service on the applicants, in the form of an affidavit, or, for lawyers, a certificate of service. Hearing requests should state the nature of the writer's interest, the reason for the request, and the issues contested. Persons who wish to be notified of a hearing may request notification by writing to the Commission's Secretary.

ADDRESSES: Secretary, Commission, 450 Fifth Street, NW., Washington, DC 20549–0609; Applicants, c/o Keith J. Rudolf, Esq., Jorden Burt LLP, 1025 Thomas Jefferson Street, NW., Washington, DC 20007.

FOR FURTHER INFORMATION CONTACT: Jean E. Minarick, Senior Counsel, at (202) 942–0527 and Annette M. Capretta, Branch Chief, at (202) 942–0564 (Division of Investment Management, Office of Investment Company Regulation).

SUPPLEMENTARY INFORMATION: The following is a summary of the application. The complete application may be obtained for a fee at the Commission's Public Reference Branch, 450 Fifth Street, NW, Washington, DC 20549–0102 (telephone (202) 942–8090).

Applicants' Representations

1. The Series Trust and the Investors Series Trust, Massachusetts business trusts, and the Variable Funds, each a Delaware limited liability company, are registered under the Act as open-end management investment companies and have one or more series (each a ''Fund'' and, together, the ''Funds''). Each of the