the status of meetings call (recording)—(301) 415–1292. Contact person for more information: R. Michelle Schroll (301) 415–1662.

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SUPPLEMENTARY INFORMATION: By a vote of 5–0 on November 27 and December 2, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Discussion of Security Issues (Closed—Ex. 1)'' be held on December 4, 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: *www.nrc.gov/what-we-do/policy-making/schedule.html.*

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

Dated: December 5, 2002.

R. Michelle Schroll,

Acting Technical Coordinator, Office of the Secretary.

[FR Doc. 02–31214 Filed 12–6–02; 1:35 pm]

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NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to Pub. L. 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Pub. L. 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, November 15, 2002, through November 29, 2002. The last biweekly notice was published on November 26, 2002 (67 FR 70762).

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, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By January 9, 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, 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

[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.

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, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. Because of continuing disruptions in delivery of mail to United States Government offices, it is requested that petitions for leave to intervene and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to (301) 415–1101 or by e-mail to *hearingdocket@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and because of continuing disruptions in delivery of mail to United States Government offices, it is requested that copies be transmitted either by means of facsimile transmission to (301) 415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, 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, 304–415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of amendment requests: November 7, 2002.

Description of amendment requests: The amendments would revise Technical Specification (TS) 3.2.4, ''Departure From Nucleate Boiling Ratio (DNBR),'' TS 3.3.1, ''Reactor Protective System (RPS) Instrumentation—Operating,'' and TS 3.3.3, ''Control Element Assembly Calculators (CEACs).'' The proposed changes are to Limiting Conditions for Operation (LCOs), LCO Actions, and LCO Surveillance Requirements. The amendments support the replacement of the Core Protection Calculator System (CPCS). The replacement CPCS will perform functionally identical safety-related algorithms as the existing CPCS, although on a newer platform, and the CPCS design function will remain unchanged. Because the replacement CPCS for each unit will be installed in refueling outages for the three units over at least a year, starting with the Unit 2 fall 2003 outage, the licensee has proposed to have the TSs contain both the current requirements and the new requirements with the phrases ''(Before CPC Upgrade)'' and ''(After CPC Upgrade)'' on the TSs to show which requirements apply to which case.

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 Core Protection Calculator System (CPCS) is being replaced due primarily to parts obsolescence. The replacement CPCS will perform functionally identical safety-related algorithms as the existing CPCS, but on a newer platform. The CPCS design function will remain unchanged.

The physical location of the replacement CPCS will be the same as the existing CPCS in the auxiliary protective cabinets. Installation will occur during refueling outages when the system is not required for service. [The] majority of the testing will be performed prior to installation.

The CPCS is not an initiator of any analyzed accident, but is used for mitigation of a large number of anticipated operational occurrences and a small number of accidents. Since the CPCS is not an accident initiator, and the replacement CPCS is functionally unchanged, the CPC replacement will not increase the probability of an accident.

The functionality of the existing CPCS safety related algorithms are replicated in the System Requirements Specification for the Common Q [Common Qualified] Core Protection Calculator System. The basic Common Q CPCS design concept was approved by NRC Safety Evaluation (SE), Acceptance For Referencing Of Topical Report CENPD–396–P, Rev. 01, ''Common Qualified Platform'' and Appendices 1, 2, 3 and 4, Rev. 01, dated August 11, 2000 (Ref. 2 [listed in the enclosure to the amendment request]), and there have been no significant functional changes to the design as presented. The requirements for response time and accuracy that are assumed in the Palo Verde Nuclear Generating Station (PVNGS) Updated Final Safety Analysis Report (UFSAR) accident analysis will continue to be met. Therefore, since the new [replacement] CPCS will be capable of performing the same safety-related functions within the same response time and accuracy as the existing CPCS, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

Response: No.

The CPCS provides a monitoring and detection function and is not an initiator for any accident. The CPCS provides Reactor Protection System (RPS) trips on Low Departure from Nucleate Boiling Ratio (DNBR) and High Local Power Density (LPD) in response to calculations involving several input variables. It also provides a Control Element Assembly Withdrawal Prohibit (CWP) signal to the Plant Protection System (PPS), and provides indication and annunciation. The CPCS performs no other plant functions, and is not used to initiate any ESF [(Engineered Safety Feature)] functions. 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 [new] CPCS is a replacement for the existing CPCS. It will retain the same safety-related functionality as the existing CPCS. The equipment will be qualified in accordance with requirements described in the Palo Verde UFSAR.

The replacement CPCS will perform functionally identical safety-related algorithms as the existing CPCS, will trip in response to the same inputs with equivalent accuracy, and will meet the same four channel separation requirements. The only significant area of difference involves the platform. The Common Q platform uses a consistent set of qualified building blocks (Advant Controllers, Flat Panel Displays, Power Supplies, and Communication Systems) that can be used for any safety system application. For Palo Verde purposes, the only application of this platform at this time will be for use as a CPCS. The new platform will include improved human factors and fault tolerance within each CPCS channel.

In summary, the replacement CPCS performs the same function as the existing CPCS, meets the qualification requirements of the existing CPCS, and meets the accuracy standards of the existing CPCS. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, APS [(the licensee)] concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of ''no significant hazards consideration'' is justified.

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

Attorney for licensee: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, PO Box 53999, Mail Station 9068, Phoenix, Arizona 85072–3999.

NRC Section Chief: Stephen Dembek.

Carolina Power & Light Company, Docket No. 50–324, Brunswick Steam Electric Plant, Unit 2, Brunswick County, North Carolina

Date of amendment request: November 7, 2002.

Description of amendment request: The proposed amendment would revise the Minimum Critical Power Ratio (MCPR) Safety Limit contained in Technical Specification 2.1.1.2 from 1.09 to 1.11 for two recirculation loop operation and from 1.10 to 1.13 for single recirculation loop operation.

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:

CP&L [Carolina Power and Light Company] has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, ''Issuance of amendment,'' as discussed below:

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

Response: No.

The MCPR Safety Limit values are calculated to ensure that greater than 99.9 percent of the fuel rods in the core avoid transition boiling during any plant operation if the safety limit is not violated. The derivation of the MCPR Safety Limit values specified in the Technical Specifications, and their use to determine cycle-specific thermal limits, has been performed using the methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (*i.e.*, GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–14–US, June 2000, which incorporates Amendment 25. Amendment 25 was approved by the NRC in a March 11, 1999, safety evaluation report. Operational MCPR limits are applied that ensure the MCPR Safety Limit is not exceeded during all modes of operation and anticipated operational occurrences.

The revised MCPR Safety Limit values do not affect the operability of any plant systems nor do these revised values compromise any fuel performance limits; therefore, the probability of fuel damage will not be increased as a result of this change.

The MCPR Safety Limit values do not impact the source term or pathways assumed in accidents previously evaluated, and there are no adverse effects on the factors contributing to offsite or onsite radiological doses. In addition, the revised MCPR Safety Limit values do not affect the performance of any equipment used to mitigate the consequences of a previously evaluated accident and do not affect setpoints that initiate protective or mitigative actions.

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

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

Response: No.

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in allowable modes of operation. The proposed revision of the MCPR Safety Limit values does not involve any facility modifications, and plant equipment will not be operated in a different manner. No new initiating events or transients will result from the revised MCPR Safety Limit values. As a result, no new failure modes are being introduced. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The margin of safety is established through the design of the plant structures, systems, and components; through the parameters within which the plant is operated; through the establishment of setpoints for actuation of equipment relied upon to respond to an event; and through margins contained within the safety analyses. The revised MCPR Safety

Limit values will not adversely impact the performance of plant structures, systems, components, and setpoints relied upon to respond to mitigate an accident or transient. The MCPR Safety Limit values are calculated to ensure that greater than 99.9 percent of the fuel rods in the core avoid transition boiling during any plant operation if the safety limit is not violated, thereby ensuring that fuel cladding integrity is maintained. The revised MCPR Safety Limit values have been calculated using NRC approved methods and procedures and preserve the existing margin to transition boiling. Based on the assurance that the fuel design criteria are being met, the revised MCPR Safety Limit values do not involve a reduction in a margin of safety.

Based on the above, CP&L has concluded that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of ''no significant hazards consideration'' is justified.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the 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 G. 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: November 14, 2002.

Description of amendment request: The proposed amendments would revise the Technical Specification Surveillance Requirement (SR) 3.3.1.3 to add a correlation slope to the formula for imbalance error. The SR is also being changed to require an adjustment of the power range channel output if the absolute value of the imbalance error is ≥2 percent rated thermal power.

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 Power Company (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures 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.

No. This change will add a correlation slope (CS) to Imbalance Error that is derived from the Power Imbalance Detector Correlation (PIDC) test performed during the cycle startup testing. The formula currently exists in the technical specification. The CS will add nuclear conservatism to the error calculation.

Since the calculation already exists and the CS adds more conservatism, this proposed change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

No. As stated above, the proposed revision adds a conservative CS to the existing error calculation. This change is bounded by all of the existing accidents and does not create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

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

No. The proposed change does not adversely affect any plant safety limits, set points, or design parameters. The change also does not adversely affect the fuel, fuel cladding, Reactor Coolant System, or containment integrity. Therefore, the proposed change does not involve a significant in a margin of safety.

Duke has concluded, based on the above, that there are no significant hazards considerations involved in this amendment request.

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.

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

Date of amendment request: October 22, 2002.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to change TS Section 5.0, ''Administrative Controls,'' to adopt Technical Specification Task Force (TSTF) –258, Revision 4. The proposed changes would: (1) Revise TS Section 5.2.2, ''Unit Staff,'' to delete the details of the staffing requirements and delete the requirements for the Shift Technical Advisor (STA) as a separate position while retaining the function, (2) revise TS Section 5.5.4, ''Radioactive Effluent Controls Program,'' to be consistent with the intent of 10 CFR Part 20, (3) revise TS Section 5.6.4, ''Monthly Operating Reports,'' to delete periodic reporting requirements for main steam safety/relief valve challenges to be consistent with Generic Letter 97–02, ''Revised Contents of the Monthly Operating Report,'' and (4) revise TS Section 5.7, ''High Radiation Area,'' in accordance with 10 CFR 20.1601(c). A new TS Section 5.3.2 would be added to incorporate regulatory definitions for the senior reactor operator (SRO) and reactor operator (RO) positions.

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

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change is an administrative clarification of existing TS requirements which clarifies and modifies administrative controls in the areas of operator staffing requirements, working hour limits, STA position, Radioactive Effluent Controls Program, periodic reporting requirements for relief valve openings, and radiological control requirements. These changes do not impact the operation, physical configuration, or function of plant equipment or systems. These TS revisions do not affect analysis inputs or mitigation for analyzed accidents and transients. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. The proposed change does not introduce any new modes of plant operation or make any changes to system setpoints. Therefore, the proposed amendment does 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 proposed change is administrative in nature and does not involve physical changes to plant structures, systems, or components (SSCs), or the manner in which these SSCs are operated, maintained, modified, tested, or inspected. The proposed change does not involve a change to any safety limit, limiting safety system setting, limiting condition for operation, or design parameters for any SSC. The proposed change does not impact any safety analysis assumptions and does not involve a change in initial conditions, system response times, or other parameters affecting any accident analysis.

For these reasons, the proposed amendment 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: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: October 22, 2002.

Description of amendment request: The proposed amendment deletes a reference to Section 2.E in Section 2.F of Facility Operating License No. NPF–21. Section 2.E requires the licensee to fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans. Section 2.E is redundant because the reporting requirements and criteria for the Physical Security Programs are specified in 10 CFR 73.71 and Appendix G of 10 CFR part 73.

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 Operating License amendment request is administrative in nature and merely deletes a duplicative and unnecessary reporting requirement. The proposed amendment deletes a reference to Operating License Section 2.E in Operating License Section 2.F. Operating License Section 2.F presently requires the Columbia Generating Station to report any violations of the requirements contained in Section 2.C (with the exception of 2.C(2)) and 2.E of the License. Operating License Section 2.E requires Columbia Generating Station to fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans. The requirement to report a violation of Section 2.E is redundant and unnecessary because the reporting requirements and criteria for the physical security program are specified in [10 CFR 73.71 and 10 CFR 73] Appendix G. This change to the Operating License has no impact on the manner in which the Columbia Generating Station is operated. No actual plant equipment or accident analyses will be affected by the proposed change. There will be no increase in radiological dose to plant workers or the public. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed Operating License amendment request is administrative in nature and merely deletes a duplicative and unnecessary reporting requirement. The proposed amendment deletes a reference to Operating License Section 2.E in Operating License Section 2.F. Operating License Section 2.F presently requires the Columbia Generating Station to report any violations of the requirements contained in Section 2.C (with the exception of 2.C(2)) and 2.E of the License. Operating License Section 2.E requires Columbia Generating Station to fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans. The requirement to report a violation of Section 2.E is redundant and unnecessary because the reporting requirements and criteria for the Physical Security Program are specified in 10 CFR 73.71 and 10 CFR 73 Appendix G. This request is administrative in nature. This change to the Operating License has no impact on the manner in which the Columbia Generating Station is operated. No actual plant equipment or accident analyses will be affected by the proposed change. No failure modes not bounded by previously evaluated accidents will be created. Therefore, the proposed amendment does 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 proposed change will not reduce a margin of safety because it has no direct effect on any safety analyses assumptions, and no adverse effect on the performance of any system, structure, or component relied upon for accident mitigation. The proposed amendment deletes a reference to Operating License Section 2.E in Operating License Section 2.F. Deletion of the reference to Section 2.E eliminates a redundant and unnecessary reporting requirement, because the reporting requirements and criteria for the physical security program are specified in 10 CFR 73.71 and 10 CFR 73 Appendix G. Additionally, there would be no effect on baseline core damage probability. 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: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: September 18, 2002.

Description of amendment request: The proposed change will revise the Technical Specifications (TS) Limiting Conditions for Operation and Administrative sections to correct or clarify certain requirements and information.

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 changes are primarily to correct word omissions, typographical errors, reflect current terminology, and make the TS consistent with other NRC [U.S. Nuclear Regulatory Commission] approved documents. These changes are all of an administrative nature and have no effect on any plant equipment or structures. Therefore, these changes do not increase the probability or consequences of an accident previously evaluated.

The proposed amendment also revises the allowed drywell-to-primary containment differential pressure limit. This limit is intended to ensure that containment conditions are consistent with safety analyses. The proposed smaller negative pressure ensures that the design assumptions for the containment will be met if and when a postulated loss of coolant [accident] (LOCA) should occur. Moving the limit in a conservative direction will not increase the probability or consequences of previously evaluated accidents.

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 do not involve a physical alteration of the plant. No new or different equipment or modes of operation are being introduced by this proposed change. Thus, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The change to the allowed drywell-to-primary containment differential pressure limit does not adversely impact the ability of the containment to perform its intended function. The establishment of a more conservative limit for this parameter ensures that the plant stays within current safety analysis and therefore, can not create the possibility of a new or different kind of accident.

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

Response: No.

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. The proposed changes are primarily administrative in nature and can not affect any safety barriers. The proposed change to the allowed drywell-to-primary containment differential pressure limit establishes a more conservative limit for a key parameter for the containment than is currently specified in the TS. The revised differential pressure limit is consistent with current assumptions of the accident analysis. 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., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi; Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana; and Entergy Operations, Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: November 6, 2002.

Description of amendment request: The proposed change will delete the content of the Appendix B, Environmental Protection Plan (Non-Radiological) (EPP), and the appropriate sections of the Facility Operating License (FOL) referring to the EPP will be modified to delete reference to the EPP.

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 EPPs are concerned with monitoring the effect that plant operations have on the environment for the purpose of protecting the environment and has no affect on any accident postulated in the Updated Final Safety Analysis Report (UFSAR). Accident probabilities or consequences are not affected in any way by the environmental monitoring and reporting required by the EPPs. The deletion of Appendix B of the FOL will not impact the design or operation of any plant system or component. The NRC [Nuclear Regulatory Commission] relies on other Federal, State, and local agencies for environmental protection regulation. No environmental protection requirements established by these other agencies are being reduced by this license amendment. The programs and reporting requirements of the EPPs do not affect the initiation or mitigation of any accidents previously analyzed.

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.

This license amendment is administrative in nature. Environmental monitoring and reporting has no affect on accident initiation. The deletion of the EPPs will not produce any changes to the design or operation of the plant. There will be no effect on the types and amounts of any effluent that will be released.

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.

This change is administrative in nature. The change in annual reporting requirements has no impact on margin of safety. Environmental Evaluations will still be performed, where necessary, on changes to plant design or operations to assess the effect on environmental protection.

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.

Attorneys for licensee: (Grand Gulf Nuclear Station, Unit 1, and Waterford Steam Electric Station, Unit 3) Nicholas S. Reynolds, Esq., Winston & Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502; and (River Bend Station, Unit 1) Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: August 16, 2002.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) 3/4.10.A, ''Refueling Interlocks'' to provide an alternative required action if the refueling interlocks became inoperable during fuel movements in the reactor vessel. The proposed amendment would also modify TS 3/4.10.D, ''Multiple Control Rod Removal.'' The proposed changes would allow fuel movements in the reactor vessel should the refueling equipment interlocks become inoperable.

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 refueling interlocks function to prevent prompt reactivity excursions during refueling. Criticality and, therefore, subsequent prompt reactivity excursions are prevented during the insertion and during control rod movement provided the other control rods in core cells containing one or more fuel assemblies are fully inserted. The refueling interlocks accomplish this by preventing loading of fuel into the core with any control rod withdrawn, by preventing withdrawal of a rod from the core during fuel loading, or preventing multiple control rod withdrawal. The proposed requirements ensure that these functions can be performed when required. Therefore, the probability of an accident previously evaluated is not significantly increased.

The refueling interlocks addressed by these specifications do not mitigate the consequences of any accident. Therefore, consequences of an accident previously evaluated are not significantly increased.

2. Does the proposed change create the possibility of a new or different kind of

accident [from] any accident previously evaluated?

Response: No.

The proposed change does not involve a change to the plant design. The refueling interlocks function to prevent prompt reactivity excursions during refueling. The proposed requirements ensure that these functions can be performed when required. As a result, the proposed changes do not affect any of the parameters or conditions that could contribute to the initiation of any new or different kind of accident. Therefore, this proposed [change] does not create the possibility of a new or different kind of accident [from] any accident previously evaluated.

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

Response: No.

The refueling interlocks function to prevent prompt reactivity excursions during refueling. Criticality and, therefore, subsequent prompt reactivity excursions are prevented during the insertion of fuel, provided all control rods are fully inserted during the fuel insertion and during control rod movement provided the other control rods in core cells containing one or more fuel assemblies are fully inserted. The refueling interlocks accomplish this by preventing loading of fuel into the core with any control rod withdrawn, by preventing withdrawal of a rod from the core during fuel loading, or preventing multiple control rod withdrawal. The proposed requirements ensure that these functions can be performed when required. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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. Andersen, Acting.

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

Date of amendment request: August 16, 2002.

Description of amendment request: The proposed amendment would delete Technical Specification (TS) 3.10.D.1.d from TS 3/4.10.D, ''Multiple Control Rod Removal,'' and the associated Surveillance Requirement 4.10.D.1.d. The proposed changes involving the deletion of this requirement would reduce the number of fuel movements or valve manipulations, thereby, increasing safety and reducing worker dose. In addition, the proposed amendment would also make an editorial change to correct a reference to TS 3.3.B.3 instead of TS 3.3.B.4 in TS 3/4.10.D.1.

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

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

Response: No.

Following the deletion of the requirement that all control rods in a 3×3 array centered on each of the control rods being removed be fully inserted and electronically or hydraulically disarmed, or have the surrounding four fuel assemblies removed from the core cell, sufficient barriers will be in place to prevent the possibility of an unacceptable reactivity excursion.

As a backup to licensee procedures and controls to prevent an unacceptable reactivity excursion, the Technical Specifications (TS) will continue to have two layers of controls to ensure that an unacceptable reactivity excursion cannot occur. The first layer of control is on the local reactivity effects of withdrawing the control rod while the second is on any potential core wide effects.

The local reactivity effects of removing the control rod are addressed by the requirement that the four fuel assemblies be removed from the core cell surrounding each control rod or control rod drive mechanism to be removed from the core and/or the reactor vessel. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core ensures withdrawal of another control rod cannot result in an unacceptable reactivity excursion.

Any potential core wide effects of removing the control rod will also continue to be controlled by the TS. The TS will continue to require control rods that are not withdrawn in accordance with 3/4.10.D remain fully inserted, the core remain sub-critical with a margin with the highest worth control rod withdrawn, and no more than one control rod can be inadvertently withdrawn. These requirements together ensure an operator error that resulted in the withdrawing of a control rod from a fueled cell would not result in an unacceptable reactivity excursion and the operator cannot withdraw a second control rod in error. Therefore, these requirements ensure that adequate [Shutdown Margin] SDM will be maintained, thereby, preventing unacceptable reactivity excursions during refueling.

In addition to these two barriers preventing an unacceptable reactivity excursion, the TS will continue to require that the source range monitors be operable. This requirement ensures that neutron monitoring information is available to the operators providing them with the information necessary to identify an unacceptable reactivity excursion is occurring and take action to terminate the event.

The control remaining provide sufficient assurance an unacceptable reactivity excursion will not occur during these activities. Therefore, the probability of an accident previously evaluated is not significantly increased.

The control being deleted did not mitigate the consequences of any accident. Therefore, consequences of an accident previously evaluated are not significantly increased.

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

Response: No.

The proposed change does not involve a change to the plant design or a new mode of equipment operation. As a result, the proposed change does not affect parameters or conditions that could contribute to the initiation of any new or different kind of accident. Therefore, this proposed [change] does not create the possibility of a new or different kind of accident [from] any accident previously evaluated.

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

Response: No.

Following the deletion of the requirement that all control rods in a 3×3 array centered on each of the control rods being removed be fully inserted and electrically or hydraulically disarmed, or have the surrounding four fuel assemblies removed from the core cell, sufficient barriers will be in place to prevent the possibility of an unacceptable reactivity excursion.

The TS will continue to have controls as a backup to licensee procedures and controls to prevent an unacceptable reactivity excursion. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core ensures withdrawal of another control rod cannot result in an unacceptable reactivity excursion. Also the TS will ensure that an operator error which results in the withdrawing of a control rod from a fueled cell will not result in an unacceptable reactivity excursion and that the operator cannot withdraw a second control rod in error.

In addition to these two barriers preventing an unacceptable reactivity excursion, the TS will continue to require that the source range monitors be operable. This requirement ensures that neutron monitoring information is available to the operators providing them with the information necessary to identify that an unacceptable reactivity excursion is occurring and take action to terminate the event.

The controls remaining provide sufficient assurance an unacceptable reactivity excursion will not occur during these activities. 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 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. Andersen, Acting.

Entergy Operations Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: October 24, 2002.

Description of amendment request: The amendment would revise Technical Specifications (TSs) relating to positive reactivity additions while in shutdown modes by clarifying TSs involving the positive reactivity additions. The proposed changes are based on Technical Specification Task Force (TSTF)–286, Revision 2, and allow for small, controlled, safe insertions of positive reactivity while in shutdown modes. In addition, two administrative-type changes are proposed.

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 (TS) changes revise actions that either require suspension of operations involving positive reactivity additions or preclude reduction in boron concentration less than the reactor coolant system (RCS). Reactivity excursions are analyzed events. The proposed changes limit positive reactivity additions into the RCS such that the required shutdown margin (SDM) or refueling boron concentration continue to be met. Reactivity changes performed during shutdown modes are currently governed by strict administrative controls. Although the proposed changes will allow procedural flexibility with regards to RCS temperature and boron concentration, these operations will still be under administrative control. The changes proposed by these amendments are within the scope and assumptions of the existing analyses.

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

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

Response: No.

The proposed TS revisions relate to positive reactivity additions while in shutdown modes of operation. Reactivity excursions are analyzed events. The operational flexibility allowed in these proposed license amendments will be performed under strict administrative controls in order to limit the potential for excessive positive reactivity addition. Although the existing procedural controls will need modification, no new or different operational failure modes will be introduced by these changes.

Additionally, implementation of these proposed changes does not require any physical plant modifications, so no new or different hardware-related failure modes are introduced. The changes proposed by these amendments are within the scope and assumptions of the existing analyses.

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

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

Response: No.

The proposed changes conform closely to the industry and NRC approved TSTF–286, Rev[ision] 2, and relate to small, controlled, safe insertions of positive reactivity additions while in shutdown modes. These changes revise actions that either require suspension of operations involving positive reactivity additions, or prohibit RCS boron concentration reduction. The proposed changes provide operational flexibility while controlling positive reactivity additions. The proposed changes provide for continued safe reactor operations and preserve the required SDM or refueling boron concentration.

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: N. S. Reynolds, Esquire, Winston & Strawn 1400 L Street NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: October 16, 2002.

Description of amendment request: The proposed amendment would revise the Completion Time for Required Action A.1 of TS 3.8.7, ''Inverters—Operating,'' from the current 24 hours for one instrument bus inverter inoperable to 14 days. The change is being proposed to support on-line maintenance of the instrument bus inverters and will have a negligible impact on plant safety. The current Completion Time for restoration of an inoperable instrument bus inverter is insufficient to support the required maintenance and post-maintenance testing windows.

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

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

The proposed action allows continued unit operation, for up to 14 days, with an inoperable instrument bus inverter. An inoperable instrument bus inverter is not considered as an initiator of any analyzed event. Extending the Completion Time for an inoperable instrument bus inverter would not have a significant impact on the frequency of occurrence for any accident previously evaluated. The proposed change will not result in changes to the plant activities associated with instrument bus inverter maintenance, but rather will allow increased flexibility in the scheduling and performance of preventive maintenance. Therefore, this change will not significantly increase the probability of occurrence of any event previously analyzed in the current Byron/Braidwood Stations' Updated Final Safety Analysis Report (UFSAR) safety analyses.

The consequences of a previously analyzed event are dependent on the initial conditions assumed in the analysis, the availability and successful functioning of equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. With an instrument bus inverter inoperable, the affected instrument bus is capable of being fed from its dedicated safety-related constant voltage transformer (CVT), which is powered from a 480 VAC Engineered Safety Feature (ESF) bus. In the event of a Loss of Offsite Power (LOOP), the affected instrument bus will experience a momentary loss of power until the associated diesel generator (DG) re-energizes the 480 VAC ESF bus. A LOOP with an inoperable instrument bus inverter (*i.e.,* instrument bus being powered by its CVT) will result in a loss of power to the associated instrument bus until the associated DG re-energizes the 480 VAC ESF bus. All instruments supplied by the instrument bus would be restored with no adverse impact to the units because no other instrument channels in the opposite train would be expected to be inoperable or in a tripped condition during this time, with the exception of routine surveillances. In the event the DG failed (*i.e.,* failed to re-energize the 480 VAC ESF bus), power could still be established to the 4 kV ESF bus by powering the 480 VAC ESF bus from the opposite unit 4 kV ESF bus cross-tie breaker. In the event of a failure to re-energize the 480 VAC ESF bus or of a CVT failure, the most significant impact on the unit is the failure of one train of ESF equipment to actuate. In this condition, the redundant train of ESF equipment will automatically actuate to mitigate the accident, and the affected unit would remain within the bounds of the accident analyses. Therefore, the request for extending the Completion Time will not significantly increase the consequences of an accident previously evaluated in the Byron/Braidwood Stations' UFSAR.

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

The proposed action does not involve physical alteration of the station. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There is no change being made to the parameters within which the units are operated. There are no setpoints at which protective or mitigative actions are initiated that are affected by this proposed action. The use of the CVT as an alternate power source for the instrument bus is consistent with the Byron and Braidwood Stations' plant designs. This proposed action will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alteration in the procedures, which ensure the unit remains within analyzed limits, is proposed, and no change is being made to procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced. The proposed action does not alter assumptions made in the safety analysis.

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?

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 or actions. There is no change in the design of the affected systems, no alteration of the setpoints at which alarms or actions are initiated, and no change in plant configuration from original design. With one of the required instrument buses being powered from the CVT, there is no significant reduction in the margin of safety. Testing of the DGs and associated electrical distribution equipment provides confidence that the DGs will start and provide power to the associated equipment in the unlikely event of a LOOP during the extended 14-day Completion Time.

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

Based on the above evaluation, we have concluded that the proposed 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 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.

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: October 10, 2002.

Description of amendment request: The proposed amendments would change technical specifications to increase the number of safety valves required to be operable from eight to nine and add surveillance requirements for the ninth safety valve.

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

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

The proposed Technical Specifications (TS) changes require an additional safety valve to be operable. The proposed change also adds the requirement to verify the lift setpoint of this additional safety valve. TS requirements that govern operability or routine testing of plant components are not assumed to be initiators of any analyzed event because these components are intended to prevent, detect, or mitigate accidents. Therefore, these changes will not involve an increase in the probability of an accident previously evaluated.

The proposed changes ensure that the reactor pressure vessel (RPV) steam dome pressure response is maintained within established limits in order to maintain the analyzed response of the RPV steam dome pressure below the safety limit for this parameter during the most severe pressurization transient. This ensures that the reactor coolant system integrity will be maintained during this transient. Thus, the proposed change does not involve an increase in the consequences of an accident previously evaluated.

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

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

The proposed changes do not affect the manner in which plant systems will be operated under normal and abnormal operating conditions. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes ensure that the RPV steam dome pressure response is maintained within established limits in order to maintain the analyzed response of the RPV steam dome pressure below the safety limit for this parameter during the most severe pressurization transient. Ensuring the safety limit is met for this transient ensures that RCS integrity will be maintained. Therefore, the proposed changes do not result in a 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 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.

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: October 28, 2002.

Description of amendment request: The proposed amendments would authorize changes to the Updated Final Safety Analysis Report (UFSAR) to address the use of cast iron components in the containment cooling service water and emergency diesel generator cooling water systems. These changes were submitted to the Nuclear Regulatory Commission (NRC) for review and approval in accordance with 10 CFR 50.59(c)(2).

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

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

The proposed changes allow for the use of cast iron materials in the Containment Cooling Service Water (CCSW) and Diesel Generator Cooling Water (DGCW) Systems at Dresden Nuclear Power Station (DNPS). The use of cast iron materials in these systems would be subject to acceptance criteria proposed for incorporation into the DNPS Updated Final Safety Analysis Report (UFSAR).

A failure in the CCSW or DGCW systems is not an initiator of any analyzed accident described in the UFSAR. Therefore, these proposed changes would not involve an increase in the probability of an accident previously evaluated. Additionally, these proposed changes would not increase the consequences of an accident previously evaluated because the proposed changes would not adversely impact structures, systems, or components. The proposed UFSAR acceptance criteria establish requirements for cast iron use that ensure the CCSW and DGCW systems would be capable of performing their intended safety-related functions of supplying cooling water to essential plant equipment, even during a design basis earthquake.

In summary, 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 allow for the use of cast iron materials in the CCSW and DGCW systems at DNPS by adding acceptance criteria to the UFSAR for such material. No other changes in requirements are being proposed. The added acceptance criteria establish requirements for cast iron that ensure the CCSW and DGCW systems would be capable of performing their safety-related functions of supplying cooling water to essential plant equipment, even during a design basis earthquake. No new failure modes are introduced by the proposed change. No new sources of energy are added. There is no change being made to the parameters within which DNPS is operated, nor do the proposed changes physically alter the plant. The proposed changes do not adversely impact the manner in which the CCSW or DGCW systems will operate under normal and abnormal operating conditions. The plant response to any single failure is not changed. The proposed changes will not alter the function demands on credited equipment. No alteration in the procedures, which ensure DNPS remains within analyzed limits, is proposed, and no change is being made to procedures relied upon to respond to an off-normal event. Therefore, these proposed changes provide an equivalent level of safety and will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The CCSW and DGCW systems are addressed in Technical Specifications (TS) Sections 3.7.1 and 3.7.2. However, the Bases of these TS sections do not discuss the codes to which the systems are designed. 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 cast iron acceptance criteria will ensure that any implied margin of safety is maintained regarding the ability of the CCSW and DGCW systems to perform their safety functions during all design basis conditions. Therefore, it is concluded that the proposed changes do not result in 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 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.

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

Date of amendment request: October 24, 2002.

Description of amendment request: The proposed amendments would revise Technical Specification 5.5.13, ''Primary Containment Leakage Rate Testing Program,'' to reflect a one-time deferral of the primary containment Type A test to no later than June 13, 2009, for Unit 1 and no later than December 7, 2008, for Unit 2.

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

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

No. The proposed changes will revise LaSalle County Station, Units 1 and 2, Technical Specification (TS) 5.5.13, ''Primary Containment Leakage Rate Testing Program'' to reflect a one-time deferral of the primary containment Type A test to no later than June 13, 2009, for Unit 1 and no later than December 7, 2008, for Unit 2. The current Type A test interval of ten years, based on past performance, would be extended on a one-time basis to 15 years from the last Type A test.

The function of the primary containment is to isolate and contain fission products released from the reactor Primary Coolant System (PCS) following a design basis Loss-of-Coolant Accident (LOCA) and to confine the postulated release of radioactive material to within limits. The test interval associated Type A testing is not a precursor of any accident previously evaluated. Type A testing does provide assurance that the LaSalle County Station primary containments will not exceed allowable leakage rate values specified in the Technical Specifications and will continue to perform their design function following an accident. The risk assessment of the proposed changes has concluded that there is an insignificant increase in total population dose rate and an insignificant increase in the conditional containment failure probability.

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

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

No. The proposed changes for a one-time extension of the Type A tests for LaSalle County Station, Units 1 and 2 will not affect the control parameters governing unit operation or the response of plant equipment to transient and accident conditions. The proposed changes do not introduce any new equipment, modes of system operation or failure mechanisms.

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

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

No. LaSalle County Station, Units 1 and 2, are General Electric BWR/5 plants with Mark II primary containments. The Mark II primary containment consists of two compartments, the drywell and the suppression chamber. The drywell has the shape of a truncated cone, and is located above the cylindrically shaped suppression chamber. The drywell floor separates the drywell and the suppression chamber. The primary containment is penetrated by access, piping and electrical penetrations.

The integrity of the primary containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRT) and the overall leak tight integrity of the primary containment is verified by a Type A integrated leak rate test (ILRT) as required by 10 CFR 50, Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.'' These tests are performed to verify the essentially leak tight characteristics of the primary containment at the design basis accident pressure. The proposed changes for a one-time extension of the Type A tests do not effect the method for Type A, B or C testing or the test acceptance criteria.

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 Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2, Beaver County, Pennsylvania

Date of amendment request: June 5, 2002, as supplemented August 19, 2002.

Description of amendment request: The requested amendments would change the plant technical specifications (TSs) to allow plant operation with the associated containment at atmospheric pressure. The plant TSs currently require the containment to be maintained at sub-atmospheric pressures when its associated unit is in operation. Minor editorial, formatting, and pagination changes will also be made as necessary to incorporate the revisions into the TSs.

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 Beaver Valley Power Station (BVPS) containments are designed to withstand the internal pressure and temperature resulting from a loss of coolant accident (LOCA), main steamline break (MSLB), feedwater line break, and a control rod ejection accident (CREA). All of these accidents have been previously analyzed in the Updated Final Safety Analysis Report (UFSAR) except the feedwater line break. This is not analyzed because the MSLB is most limiting. The effect on containment pressure and temperature due to a CREA is bounded by a LOCA, since a CREA is modeled as a small break LOCA. The probability of occurrence for these accidents is independent of the type of containment. Therefore a change from a subatmospheric to an atmospheric containment will not increase the probability of these accidents.

The revised containment integrity analysis demonstrates that the pressures and temperatures associated with the applicable design basis accidents identified above are within the existing containment design limits. From a containment integrity viewpoint, the limiting design basis accidents (DBA) presently are the MSLB for Unit 1 and the LOCA for Unit 2. Following the conversion to an atmospheric containment, the limiting DBA will be the MSLB for both units. The effects of the proposed changes on plant structures, systems and components (SSC) have been evaluated and verify that the capability of the SSCs to perform their design functions will be retained following approval of the proposed changes. The revised radiological analysis reflects a selective application of the Alternative Source Term (AST) of Regulatory Guide 1.183, ''Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,'' and incorporation of the ARCON96 methodology for on-site atmospheric dispersion factors. The revised radiological analysis concludes that normal operation of the BVPS units with atmospheric containments will not impact either unit's compliance with the operator exposure limits set forth in 10CFR20, or with the public exposure limits set forth by 10CFR50, Appendix I.

For accident conditions, the proposed changes will potentially impact the reported dose consequences of the LOCA, CREA and MSLB for both BVPS units, and the locked rotor accident (LRA) for BVPS Unit 1. The radiological consequences of the remaining design bases accidents are not adversely impacted by the proposed changes.

The revised radiological analysis concludes that site boundary and control room dose consequences of the LOCA and the CREA remain within the regulatory requirements of 10CFR50.67, as supplemented by Regulatory Guide 1.183. It also concludes that the control room doses for the MSLB for both BVPS units, and LRA for BVPS Unit 1 will continue to remain within the regulatory limits provided in SRP 6.4 [NUREG–0800, ''Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,'' section 6.4].

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 design basis accidents, which could be adversely affected by the proposed changes, have been reanalyzed. These analyses demonstrate that all acceptance criteria have been satisfied. The revised containment integrity analysis demonstrates that the containment will not be subjected to temperatures or pressures that are beyond its design limits. Converting to an atmospheric containment will not result in any new or different kind of accidents because no new accident initiators will be introduced.

Changes to instrumentation setpoints, system flow rates, surveillance requirements, and the elimination of certain operability requirements will not have any [effect] that could create the possibility of a new or different type of accident since none of these changes would result in any changes to the manner in which the affected equipment is operated. 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 margin of safety attributed to the containment involves both the pressures and temperatures the containment is subjected to following a DBA, and the on-site and offsite dose consequences associated with normal and post DBA operations.

The revised containment integrity analysis conducted to support the proposed changes demonstrate that the containment peak pressure and temperature following a DBA will not exceed the containments' design limits. Since the containment design limits are not exceeded, the existing margin of safety between these limits and the containment failure limits is not reduced.

The revised radiological analysis concludes that the existing dose consequence margin of safety is not significantly reduced. 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, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2, Beaver County, Pennsylvania

Date of amendment request: October 31, 2002.

Description of amendment request: The proposed amendments would revise the Beaver Valley Technical Specifications (TS) to allow extending the Type A Containment Integrated Leak Rate Test (ILRT) interval from 10 years to 15 years on a one-time basis.

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 does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change allows a one-time extension to the current surveillance interval for the Type A Containment Integrated Leak Rate Test (ILRT). The current test interval of ten years, based on performance history, would be extended on a one-time basis to 15 years from the last Type A test. The proposed change will not result in a significant increase in the risk of plant operation. The risk analysis was performed in accordance with Regulatory Guide 1.174 and shows that the increase in total plant risk due to the extended ILRT interval is 0.005 percent (Unit 1) and 0.02 percent (Unit 2). The delta-large early release frequency (LERF) is 1.91E–9 /yr (Unit 1) and 1.35E–9 /yr (Unit 2) when the test interval is increased from 10 to 15 years. These delta-LERF values meet the Regulatory Guide 1.174 acceptance criterion of less than 1.0E–07 per year for LERF. The proposed extension to Type A testing does not increase the probability of an accident previously evaluated, since the containment Type A test does not involve any modifications, nor a change in the way that any plant structures, systems or components (SSC) function, and does not involve an activity that could lead to equipment failure or accident initiation. The proposed extension of the test interval does not involve a significant increase in the consequences of an accident, since the study documented in NUREG–1493, has found that generically, very few potential leak paths are not identified with Type B and C tests. NUREG–1493 concluded that an increase in the Type A test interval to twenty years resulted in an imperceptible increase in risk. Containment testing and inspection provide a high degree of assurance that the containment will not degrade in a manner only detectable by Type A testing. Inspections required by the ASME Code and the Maintenance Rule are performed in order to identify indications of containment degradation that could affect leak tightness. Type B and C testing requirements and intervals required by 10 CFR 50 Appendix J are not affected by this proposed extension to the Type A test interval, and will identify any potential openings in containment penetrations that would otherwise require a Type A test. The increase in risk of the proposed change, as measured by the change in LERF is within the acceptance criterion of Regulatory Guide 1.174, therefore there will not be a significant increase in the consequences of any accidents.

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

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

Response: No.

The proposed change does not result in operation of the units in a way that would create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed extension to Type A testing does not create a new or different type of accident because no physical modifications are being made, and no compensatory measures are being imposed that could potentially lead to a failure. There are no changes to unit operation that could introduce a new failure mode or create a new or different kind of accident. The proposed change only allows a one-time extension to the current interval for Type A testing and does not change the implementation aspects of the subsequent test.

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 will not result in a significant reduction in a margin of safety. The proposed change is for a one-time extension to the current interval for Type A testing. The current test interval of ten years, based on historical performance, will be extended on a one-time basis to 15 years from the last Type A test. The NUREG–1493 study of the effects of extending the Type A test interval out to 20 years concluded that there is an imperceptible increase in plant risk. Additionally, the extended test interval will have a minimal effect on plant risk, since Type B and C testing detect over 95% of potential leakage paths. The plant specific risk analysis determined results that are consistent with the conclusions of NUREG–1493. The overall increase in the risk contribution due to the proposed change was determined to be 0.005 percent (Unit 1) and 0.02 percent (Unit 2). The delta-LERF is 1.91E–9/yr (Unit 1) and 1.35E–9/yr (Unit 2) when the test interval is increased from 10 to 15 years. The calculated impact on risk is insignificant, and meets the acceptance criterion of Regulatory Guide 1.174.

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 4, 2002.

Description of amendment request: The proposed amendment proposes a revision of pressure/temperature (P/T) limit curves for non-nuclear heatup/cooldown, core critical operation, and pressure testing for reactor coolant systems (RCSs); including an exemption request pursuant to 10 CFR 50.60(b).

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 P/T limit curves are based upon the use of an alternate material fracture toughness curve and the use of an NRC-approved methodology for calculation of neutron fluence. The proposed RCS P/T limit curves are valid through 22 Effective Full-Power Years (EFPY) and 32 EFPY.

The American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel (B&PV) Code Case N–640 permits the use of K_{Ic} as defined in ASME B&PV Code, Section XI, Appendix A, Figure A–4200–1 instead of K_{Ia} as defined in ASME B&PV Code, Section XI, Appendix G, Figure G–2210–1. The use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P/T limit curves is more technically correct than the K_{Ia} curve. The K_{Ic} curve models the slow heatup and cooldown processes that a Reactor Pressure Vessel (RPV) normally undergoes. These slow heatup and cooldown limits are enforced through the use of the PNPP [Perry Nuclear Power Plant] Technical Specification 3.4.11, "RCS Pressure and Temperature (P/T) Limits." Surveillance Requirement 3.4.11.1 states that heatup and cooldown rates will be ≤100 °F in any one hour period. The use of the K_{Ic} curve is applicable to PNPP and is inconsistent with the ASME B&PV. Therefore, the use of K_{Ic} will provide an adequate margin of safety to protect against potential RPV failure.

NRC [Nuclear Regulatory Commission] regulations require the vessel material transition temperature be adjusted to account for the effects of neutron radiation. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," provides a methodology for calculating the neutron fluence, while Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," provides the guidance for calculating the adjusted transition temperature using the fluence factor. The methodologies satisfy the requirements of 10 CFR [part] 50, Appendices G and H, and General Design Criteria 31, "Fracture Prevention of Reactor Coolant Pressure Boundary." The methodologies used to develop the proposed P/T limit curves satisfy the requirements of the regulations.

The predicted lowest upper shelf energy at 32 EFPY was greater than the minimum of 50 ft-lbs required by 10 CFR [part] 50, Appendix G. The adjusted reference temperature for the limiting material was less than the 200 °F limit required by Regulatory Guide 1.99, Revision 2. Therefore, the integrity of the RCS has been maintained. As such, the proposed curves ensure that adequate reactor vessel safety margins against nonductile failure exist during normal operation, anticipated operational occurrences, and hydrostatic testing. There are no plant modifications associated with these changes. Thus, the proposed changes do not involve a significant increase in the probability of occurrence of an accident previously evaluated.

The proposed changes do not adversely affect the integrity of the reactor vessel. Hence, the function of the reactor vessel to act as a radiological barrier during an accident is not affected. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed P/T limit curves are based upon the use of an alternate material fracture toughness curve and the use of an NRC-approved methodology for calculation of neutron fluence.

The ASME B&PV Code Case N–640 permits the use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P/T limit curves. The K_{Ic} curve models the slow heatup and cooldown processes that a RPV normally undergoes. These slow heatup and cooldown limits are enforced through the use of the PNPP Technical Specifications. Therefore, the use of K_{Ic} will provide an adequate margin of safety to protect against potential RPV failure.

NRC regulations require the vessel material transition temperature be adjusted to account for the effects of neutron radiation. The methodologies used to develop the proposed P/T limit curves satisfy the requirements of the regulations. The predicted lowest upper shelf energy at 32 EFPY was greater than the minimum of 50 ft-lbs required by 10 CFR [part] 50, Appendix G. The adjusted reference temperature for the limiting material was less than the 200 °F limit required by Regulatory Guide 1.99, Revision 2. Therefore, the integrity of the RCS has been maintained. As such, the proposed curves ensure that adequate reactor vessel safety margins against nonductile failure exist during normal operation, anticipated operational occurrences, and hydrostatic testing.

There are no plant modifications associated with these changes.

The proposed changes to the P/T limit curves do not affect the assumed accident performance of any structure, system, or component previously evaluated. The proposed changes do not introduce any new modes of system operation or failure mechanisms. 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 will not involve a significant reduction in the margin of safety.

NRC regulations require that P/T limits provide an adequate margin of safety to the conditions at which brittle fracture may occur. These regulations are set forth in 10 CFR [Part] 50, Appendix A, General Design Criteria (GDC) 31, and 10 CFR [Part] 50, Appendices G and H. Regulatory Guides 1.99 and 1.190 provide guidance for the compliance of GDC 31 and Appendices G and H. The appendices reference the requirements and guidance of ASME B&PV Code, Section XI, Appendix G for the development of P/T limit curves. The methodologies described within the regulatory guides and the ASME Code will provide P/T limit curves with the requisite margin against brittle fracture. The proposed P/T limit curves are based on these methodologies as modified by application of ASME Code Case N–640.

Although the code case proposes a change to a requirement contained in ASME, Section XI, Appendix G, the alternative allowed by Code Case N–640 is based upon industry experience gained since the inception of 10 CFR [Part] 50, Appendix G. The more appropriate assumptions and provisions allowed by the code case maintain a margin of safety that is consistent with the intent of 10 CFR [Part] 50, Appendices G and H. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: October 11, 2002.

Description of amendment request: The proposed amendment would change Technical Specification (TS) 3.4.9.1, ''Reactor Coolant System [RCS]—Pressure/Temperature Limits'' and TS 3.4.9.3, ''Reactor Coolant System—Overpressure Protection Systems'' and their associated Bases sections. Specifically, the proposed changes will replace TS Figure 3.4–2, ''Reactor Coolant System Heatup Limitations,'' Figure 3.4–3, ''Reactor Coolant System Cooldown Limitations,'' and Figure 3.4–4, ''RCS Cold Overpressure Protection Setpoints,'' to allow operation to 20 Effective Full Power Years (EFPY). The proposed change to TS 3.4.9.3 will also revise the Cold Overpressure Protection System arming temperature from 329°F to 290°F to reflect the higher allowable low temperature overpressure protection pressure limit afforded by the use of ASME Code Case N–641.

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 TS 3.4.9.1 and TS 3.4.9.3 do not result in a condition where the design, material, and construction standards that were applicable prior to the proposed changes are altered. The probability of occurrence of an accident previously evaluated for Seabrook Station is not altered by the proposed amendment to the TSs. The accidents remain the same as currently analyzed in the UFSAR [Updated Final Safety Analysis Report] as a result of changes to the P/T limits as well as those for Cold Overpressure Mitigation System (COMS). The new P/T limits are based on NRC [Nuclear Regulatory Commission] accepted methodology along with [the] American Society of Mechanical Engineers (ASME) Code alternative methodology. An exemption request to allow use of the alternative ASME methodology is included as part of this LAR [License Amendment Request]. The proposed COMS setpoint limit based on the revised P/T limits satisfies the criteria specified in the alternative ASME methodology and 10 CFR part 50 Appendix G closure head/vessel flange region pressure limit criteria. The proposed changes do not impact the integrity of the reactor coolant pressure boundary (RCPB) *i.e.* there is no change to the operating pressure, materials, system loadings, etc., as a result of this change. In addition, there is no increase in the potential for the occurrence of a loss of coolant accident. The probability of any design basis accident is not affected by this change, nor are the consequences of any design basis accident (DBA) affected by this proposed change. The proposed P/T limit curves and the COMS limits are not considered to be an initiator or contributor to any accident currently, evaluated in the Seabrook Station UFSAR. These new limits ensure the long term structural integrity of the RCPB.

Fracture toughness test data are obtained from beltline material specimens contained in surveillance capsules that are periodically withdrawn from the reactor vessel. This data allows determination of time conditions under which the vessel can be operated with adequate safety margins against non-ductile fracture throughout its service life. The second Seabrook Station surveillance capsule was removed from the reactor vessel after completion of Operating Cycle No. 5 in May 1997 and was analyzed to predict the fracture toughness requirements using projected neutron fluence calculations. For each analyzed transient and steady state condition, the allowable pressure is determined as a function of reactor coolant temperature considering postulated flaws in the reactor vessel beltline region material. The predicted radiation induced ΔRT_{NDT} was calculated using the respective reactor vessel beltline materials copper and nickel contents and the neutron fluence predicted for 20 EFPY. The RT_{NDT} and, accordingly, the operating limits for Seabrook Station were adjusted to account for the effects of irradiation on the fracture toughness of the reactor vessel beltline materials. Therefore, new operating limits are established which are represented in the revised operating curves for heatup/cooldown, criticality and inservice hydrostatic testing contained in the technical specifications. The proposed P/T limit curves and COMS setpoint limits are not considered to be an initiator or contributor to any accident currently evaluated in the Seabrook Station UFSAR.

Therefore based on the above discussion, it is concluded that the proposed revisions to TS 3.4.9.1 and TS 3.4.9.3 do not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed changes to the P/T and COMS limits will not create a new accident scenario. The requirements to have P/T and COMS protection are part of the licensing basis for Seabrook Station. The proposed technical specification amendment reflects the change in reactor vessel material properties as determined by evaluation of the most recently withdrawn surveillance capsule. Based on the surveillance capsule data, the adjusted RT_{NDT} values for the plate and weld material were within the two standard deviations of Regulatory Guide 1.99, Revision 2 predictions. As all the requisite criteria of Regulatory Guide 1.99, Revision 2 was satisfied, it was concluded that the surveillance data was credible and the beltline material was responding as empirically predicted. The new P/T limits are based on NRC accepted methodology along with American Society of Mechanical Engineers (ASME) Code alternative methodology. An exemption request to allow use of the alternative ASME methodology is included as part of this LAR. The proposed COMS setpoint limit based on the revised P/T limits satisfies the criteria specified in the alternative ASME methodology and 10 CFR part 50 Appendix G closure head/vessel flange region pressure limit criteria. The proposed changes will not alter the way any structure, system or component functions, and will not significantly alter the manner in which the plant is operated. There will be no adverse effect on plant operation or accident mitigation equipment.

Since no new failure modes are created by the proposed revisions to TS 3.4.9.1 and TS 3.4.9.3, this change does not create the possibility of a new or different kind of accident from any that was previously evaluated.

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

The existing P/T and COMS limit curves in the technical specifications are reaching their expiration for the number of years at effective full power operation. The revision of the P/T limits and COMS will ensure that Seabrook Station continues to operate within the operating limits allowed by 10 CFR 50.60 and the ASME Code. The material properties used in the development of the revised limit curves are based on the evaluation of the most recently withdrawn surveillance capsule. The application of ASME Code Case N–641 presents alternative methods for calculating P/T and COMS temperature and pressure limits in lieu of those established in ASME Section XI, Appendix G–2215. This ASME Code alternative allows analysis features that are less restrictive than those associated with previous methodologies, however these features remain conservative with respect to the requirements delineated ASME Section XI. Therefore it is concluded that the revised P/T and COMS limit curves proposed by this technical specification amendment still provide sufficient margin to preclude non-ductile fracture of the reactor vessel.

Thus, it is concluded that these proposed revisions to TS 3.4.9.1 and TS 3.4.9.3 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief (Acting): James W. Andersen.

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

Date of amendment request: October 11, 2002.

Description of amendment request: The proposed amendment would relocate Technical Specifications (TSs) 3.1.2.1, ''Reactivity Control Systems-Borations Systems-Flow Paths-Shutdown;'' 3.1.2.2, ''Reactivity Control Systems-Boration Systems-Flow Paths-Operating;'' 3.1.2.3, ''Reactivity Control Systems-Boration Systems-Charging Pumps-Shutdown;'' 3.1.2.4, ''Reactivity Control Systems-Boration Systems-Charging Pumps-Operating;'' 3.1.2.5, ''Reactivity Control Systems-Boration Systems-Borated Water Sources-Shutdown;'' 3.1.2.6, ''Reactivity Control Systems-Boration Systems-Borated Water Sources-Operating;'' and 3.4.7, ''Reactor Coolant System-Chemistry,'' to the Seabrook Station Technical Requirements Manual (SSTR) and would revise TS 3.1.2.7, ''Reactivity Control Systems-Boration Systems-Isolation of Unborated Water Sources-Shutdown.'' The proposed amendment would also revise TSs 3.4.1.2, ''Reactor Coolant System-Reactor Coolant Loops and Coolant Recirculation-Hot Standby,'' 3.4.3 ''Reactor Coolant System-Pressurizer,'' 3.4.7, ''Reactor Coolant System-Chemistry,'' and 3.9.2, ''Refueling Operations-Instrumentation,'' to adopt a portion of NUREG–1431, Revision 2, ''Standard Technical Specifications, Westinghouse Plants,'' involving a wording revision to more closely match Standard Technical Specifications. The revision to TS 3/4.9.2 would also involve surveillance changes. The associated Bases would also be modified as a result of the proposed changes.

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 TS changes propose the relocation of the boration subsystem and chemistry requirements to a licensee-controlled document. The relocation of these requirements will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

The TS changes propose the modification of the TS for ''Isolation of Unborated Water Sources—Shutdown.'' Only the demineralizers that are intended to deborate the Reactor Coolant System will need to be isolated in MODE 4, 5, or 6. Administrative controls, currently in use for the operation of the Boron Thermal Regeneration System and replenishment of demineralizer resin in the Chemical Volume and Control System, will be used to minimize the affects of an inadvertent dilution due to operation of the demineralizers. The Seabrook Station Updated Final Safety Analysis currently includes a boron dilution event analysis for each MODE of operation. Use of these administrative controls will ensure that the operation of the BTRS [Boron Thermal Regeneration System] is bounded by the boron dilution analysis. Therefore, the modification of the TS requirement will not increase the probability or consequences of an accident previously evaluated.

The TS changes propose to change the source range flux monitor requirements in MODE 6. The proposed change does not significantly affect the operability of the associated equipment. The source range neutron flux monitors are components not assumed to be initiators of analyzed events. Therefore, the change in the TS requirement for the source range instrumentation in MODE 6 will not increase the probability or consequences of an accident previously evaluated.

The additional proposed changes to the TS that will standardize terminology, relocate information to the Bases, remove extraneous information, modify the requirements to prevent rod withdrawal for operational flexibility, and make minor format changes will not result in any technical changes to the current requirements. Therefore, these additional proposed changes will not increase 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 previously evaluated.

The proposed changes to the TSs do not impact any system or component that could cause an accident, nor will it alter the plant configuration or require any unusual operator actions, nor will it alter the way any structure, system, or component functions. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed TS changes associated with the relocation of the boration subsystem and chemistry requirements to a licensee-controlled document will not result in a significant reduction in a margin of safety.

The proposed TS changes associated with the modification of the TS for ''Isolation of Unborated Water Sources—Shutdown,'' are consistent with the requirements contained in the Seabrook Station Updated Final Safety Analysis which currently includes a boron dilution event analysis for each MODE of operation. The changes result in operation within the parameters specified by the analysis. Therefore, the modification of the TS requirement will not result in a significant reduction in a margin of safety.

The proposed TS changes associated with the source range flux monitor do not significantly affect the operability of the associated equipment. Therefore, the change in the TS requirement for the source range instrumentation will not result in a significant reduction in a margin of safety.

The additional proposed changes to the TSs that will standardize terminology, relocate information to the Bases, remove extraneous information, modify requirements to prevent rod withdrawal for operational flexibility, and make minor format changes will not result in any technical changes to the current requirements. Therefore, these additional changes will not result in 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: Mr. M. S. Ross, Florida Power & Light Company, PO Box 14000, Juno Beach, FL 33408–0420.

NRC Section Chief (Acting): James W. Andersen.

Florida Power and Light Company (FPL), *et al.*, Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: October 23, 2002.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) Section 5.6, ''Design Features—Fuel Storage,'' to include the design of a new cask pit spent fuel storage rack for each unit to increase the allowable spent fuel wet storage capacity at both units and include the description of Boral™ as the neutron absorbing material used in the new cask pit storage racks. The proposal also revises the spent fuel pool (SFP) thermal-hydraulic analyses for core offload times of 120 hours after reactor shutdown and for a partial core offload as the normal offload condition. In addition the proposal includes a change in FPL's commitments regarding the Unit 2 spent fuel cooling system design basis described in the Updated Final Safety Analysis Report (UFSAR). A current UFSAR commitment regarding the Unit 2 peak SFP temperature limit during full core offloads with minimum SFP cooling will be replaced with a new design basis.

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. Would operation of the facility in accordance with the proposed amendments involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed changes to increase the spent fuel storage capacity with cask pit racks were evaluated for impact on the following previously evaluated events:

a. A fuel handling accident (FHA),
b. A heavy load drop into the cask pit,
c. A loss of SFP cooling,
d. A stored fuel criticality event,
e. A seismic event.

The probability of a fuel handling accident is not significantly increased by the proposed changes, because the same equipment (*e.g.*, the spent fuel handling crane) and procedures will be used to handle fuel assemblies and the frequency of fuel movement will be essentially the same, with or without cask pit racks. The FHA radiological consequences are not significantly increased because the source term of a single fuel assembly will remain unchanged, and the cask pit racks will be installed at the same water depth as the existing SFP racks, with the same iodine decontamination factors assumed in the FHA analysis. The structural consequences of dropping a fuel assembly on a cask pit rack were also found to be no more severe than those in the current FHA analysis.

The probability and consequences of a heavy load drop of the cask pit rack or its platform are bounded by the existing cask drop analyses, because a fuel transfer cask is much heavier than either the empty rack or platform, and cask handling will be a more frequent operation in the future than cask pit rack installation and removal. The cask pit rack will be removed prior to any cask handling operations, such that a cask drop scenario onto a cask pit rack loaded with fuel is not credible. Therefore, the probability and the consequences of a heavy load drop in the cask pit are not significantly increased.

The probability of a loss of SFP cooling is unaffected and its consequences are not significantly increased with cask pit racks installed. With the cask pit rack installed, loss of forced cooling results in a sufficient time-to-boil for the operator to recognize the condition and establish SFP makeup to compensate for water lost due to pool bulk boiling, and thereby maintain a sufficient water blanket over the stored spent fuel.

The probability and consequences of a stored fuel criticality event are not increased by the addition of a cask pit rack. The reactivity analysis for the new racks demonstrates that reactivity remains subcritical (below 0.95) for the worst-case fuel mispositioning event, without credit for soluble boron. The probability of a seismic event is unaffected and its consequences are not significantly increased with cask pit racks installed, because the structural analysis of the new racks demonstrates that the fuel storage function of the rack is unimpaired by loading combinations including seismic motion, and there is no adverse seismic-induced interaction between the rack and adjacent structures.

Based on the above, it is concluded that the proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Would operation of the facility in accordance with the proposed amendments create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed changes to add a cask pit rack to each unit do not alter the operating requirements of the plant or of the equipment credited in the mitigation of design basis accidents, nor do the proposed changes affect any of the important parameters required to ensure the safe storage of spent fuel. A new rack material (Boral™) is introduced into the pool under these changes, but based on its operating history in SFPs, there are no mechanisms that create a new or different kind of accident. The potential for dropping the new rack or its platform during installation or removal is bounded by the existing analysis for dropping a spent fuel transfer cask into the cask pit. The same equipment (*e.g.*, the spent fuel handling crane) and procedures will be used to handle fuel assemblies for the new cask pit racks as are used for existing spent fuel storage. The fuel storage configuration in the new racks will be similar to the configuration in the existing SFP storage racks, and a fuel drop or mispositioning event in the new racks does not represent a new or different kind of accident from fuel handling and mispositioning events previously evaluated. Therefore, the proposed amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Would operation of the facility in accordance with the proposed amendments involve a significant reduction in a margin of safety?

No. The effect of the proposed changes on current margins of safety were evaluated for spent fuel storage functionality and criticality, spent fuel and SFP cooling, and SFP/cask pit structural integrity. The design of the new racks uses proven technology which preserves the proper safety margins for spent fuel storage to provide a coolable and subcritical geometry under both normal and abnormal/accident conditions. The design complies with current regulatory guidelines and the ANSI [American National Standards Institute] standards, including 10 CFR 50 Appendix A GDC [General Design Criterion] 62, NUREG–0800 Section 9.1.2, the OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, Regulatory Guide 1.13, and ANSI/ANS [American Nuclear Society] 8.17. Handling the racks and platforms in accordance with the defense-in-depth approach of NUREG–0612 with temporary lift items designed to ANSI N14.6 preserves the proper margin of safety to preclude a heavy load drop in the cask pit.

The proposed SFP cooling system design basis is consistent with the regulatory guidance in NRC Standard Review Plan Section 9.1.3 for SFP temperature limits during normal and abnormal core offload conditions. The rack and SFP thermal hydraulic analyses demonstrate that the proposed SFP cooling system design basis is met, and that no bulk boiling will occur in the new rack or SFP with minimum cooling available. A loss of SFP cooling will allow sufficient time for operators to identify the condition and initiate makeup flow or restore cooling to preserve fuel cooling capability.

The new rack criticality analyses demonstrate that the subcriticality safety margin is maintained below 0.95 under all conditions, without credit for soluble boron. The structural analyses for the new racks and adjacent structures show that the rack and surrounding structures are unimpaired by loading combinations during seismic motion, and there is no adverse seismic-induced interaction between the rack and adjacent structures. Based on these evaluations, operating the facility with the proposed amendments does not involve a significant reduction in any margin of safety.

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

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

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: September 26, 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 September 26, 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: Mr. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: November 15, 2002.

Description of amendment request: The licensee proposes to revise the reactor coolant system pressure-temperature (P–T) limit curves and associated limit tables specified in Section 3/4.2.2, ''Minimum Reactor

Vessel Temperature for Pressurization,'' of the Technical Specifications (TSs). The P–T limit curves and tabular listing of P–T limit values contained in the revised figures and tables are based, in part, on an alternative methodology and will be valid for 28 effective full-power years. The alternative methodology has been endorsed by the American Society of Mechanical Engineers.

The associated licensee-controlled TSs Bases pages would also be changed to reflect the above TS changes.

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 three standards of 10 CFR 50.92(c). The NRC staff's analysis is presented below:

The first standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes, if approved by the Nuclear Regulatory Commission (NRC), will be made in a manner such that conservatism is maintained through compliance with applicable NRC regulations and guidance. No hardware design change is involved with the proposed amendment, thus there will be no adverse effect on the functional performance of any plant structure, system, or component (SSC). All SSCs will continue to perform their design functions with no decrease in their capabilities to mitigate the consequences of postulated accidents. P–T limit curves were not previously factored into the probability of accidents, nor were they factored into scenarios of previously analyzed accidents. Accordingly, the revised P–T limit curves and tabular listing of P–T limit values will lead to no increase in the consequences of an accident previously evaluated, and no increase of the probability of an accident previously evaluated.

The second standard requires that operation of the unit in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment is not the result of a hardware design change, nor does it lead to the need for a hardware design change. There is no change in the methods the unit is operated. As a result, all SSCs will continue to perform as previously analyzed by the licensee, and previously evaluated and accepted by the NRC staff. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

The third standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant reduction in a margin of safety. Since the licensee did not propose to exceed or alter a design basis or safety limit, the proposed amendment will not affect in any way the performance characteristics and intended functions of any SSC. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Richard J. Laufer.

PSEG Nuclear LLC, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: July 25, 2002, as supplemented October 21, 2002.

Description of amendment request: The amendment would modify Technical Specification (TS) requirements for missed surveillance tests in TS 4.0.3 using the Consolidated Line Item Improvement Program, modify TS 4.0.1 to be consistent with the Standard Technical Specifications (STS), and incorporate a TS Bases Control Program in Section 6.0 in accordance with the STS.

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

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

Specification 4.0.3

The proposed change relaxes the time allowed to perform a missed Surveillance. The time between Surveillances is not an initiator to 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.

Specification 4.0.1

The proposed additional requirement equating failure to meet a surveillance with failure to meet the LCO [limiting condition for operation] is consistent with current interpretation of the technical specifications. This change, along with relocation and rewording of existing requirements from Specification 4.0.3, are administrative in nature and do not adversely affect accident initiators, design functions, facility configuration or the manner of operation or control. The ability of structures, systems and components to perform their intended function remains unaffected.

Bases Control Program

The proposed change to adopt a Technical Specification Bases Control Program is also administrative in nature and does not adversely affect accident initiators, design functions, facility configuration or the manner of operation or control. The ability of structures, systems or components to perform their intended function remains unaffected. Future changes to the TS Bases will continue to be administratively controlled in accordance with the requirements of 10 CFR 50.59.

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

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

None of the three proposed changes involves a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Specification 4.0.3

The relaxed 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 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.

Specification 4.0.1

The proposed changes to TS 4.0.1, including relocation and rewording of existing requirements from Specification 4.0.3, are administrative in nature and do not reduce the level of programmatic or procedural controls associated with the Surveillance Requirements. There are no substantive differences in meaning or intent between the existing specifications and the corresponding STS requirements. Further, these changes have no impact on equipment design, configuration, analytical basis, setpoints or operation.

Bases Control Program

The proposed change to adopt a Technical Specification Bases Control Program is also administrative in nature and does not reduce the level of programmatic or procedural controls associated with the Bases. There is no impact on equipment design, configuration, analytical basis, setpoints or operation.

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.

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: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, PO Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James Andersen, Acting.

PSEG Nuclear LLC, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: October 23, 2002.

Description of amendment request: The proposed change updates the reference to 10 CFR 20.203 with the corresponding reference to 10 CFR 20.1601.

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

The proposed changes do not affect accident initiators or precursors and do not alter the design assumptions, conditions, configuration of the facility, or manner in which the plant is operated. The proposed changes do not alter or prevent the ability of structures, systems, or components to perform their intended safety function to mitigate the consequences of an initiating event within the acceptance limits assumed in the UFSAR [Updated Final Safety Analysis Report]. The proposed changes are administrative in nature. Technical Specification (TS) 6.12 will be updated to include the new 10 CFR 20 (effective 06/20/91) requirements. The proposed changes do not alter the conditions or assumptions in any of the previous accident analyses, and as a result, the radiological consequences associated with these analyses remain unchanged.

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

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

The proposed changes do not alter the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated.

The proposed changes are administrative in nature and the relocated procedural details do not change the level of programmatic controls and procedural details. Accordingly, the proposed changes do not create any new failure modes or limiting single failures associated with a plant structure, system, or component important to safety. Also, there will be no change in the types or increase in the amounts of any effluents released offsite.

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

The proposed changes do not impact equipment design or operation, nor do the changes affect any TS safety limits or safety system settings that could adversely affect plant safety. The proposed changes are administrative in nature. Technical Specification (TS) 6.12 will be updated to include the new 10 CFR 20 requirements (effective 06/20/91) and are in conformance with NUREG–1433 [Standard Technical Specifications General Electric Plants, BWR 4]. Furthermore, the proposed changes do not result in a change in the types or an increase in the amounts of any effluents released offsite.

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

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

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, PO Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James Andersen, Acting.

PSEG Nuclear LLC, Docket No. 50–354, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: September 20, 2002.

Description of amendment request: The proposed amendment will add new limiting conditions for operation for fuel storage pool boron concentration, fuel assembly storage in the spent fuel pool, relocate requirements for spent fuel storage, revise existing Technical Specification (TS) 3/4.9.1 for boron concentration during refueling operations, and revise existing administrative controls associated with the Core Operating Limits Report described in TS 6.9.1.9.

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 postulated accidents are basically of three types. The first type of postulated accident is an abnormal location of a fuel assembly, the second type of postulated accident is associated with lateral rack movement, and the third type of postulated accident is a dropped fuel assembly on the top of the rack. The dropped fuel assembly and the lateral rack movement have been previously shown to have negligible reactivity effects (<0.0001 [delta k]). The misplacement of a fuel assembly could have a small positive reactivity effect, however, the negative reactivity effect of a minimum soluble boron concentration of 600 ppm [parts per million] compensates for the increased reactivity caused by any of the postulated accident scenarios.

There is no increase in the probability of the accidental misloading of irradiated fuel assemblies into the spent fuel pool racks when considering the presence of soluble boron in the pool water for criticality control. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures and will be in accordance with the Technical Specification (TS) spent fuel rack storage configuration limitations.

There is no increase in the consequences of the accidental misloading of irradiated fuel assemblies into the spent fuel pool racks because criticality analyses demonstrate that the pool will remain subcritical following an accidental misloading if the pool contains an adequate boron concentration. This has been previously evaluated in the Safety Evaluation by the Office of Nuclear Reactor Regulation related to Amendment Nos[.] 151 and 131 to Facility Operating Licenses DPR–70 and DPR–75 for the Salem Nuclear Generating Station Units 1 and 2, dated May 4, 1994 (Spent Fuel Reracking, TAC [technical assignment control] NOS. M85797 and M85798). The proposed TS limitations will ensure that an adequate spent fuel pool boron concentration will be maintained.

The proposed change will revise the Salem Generating Station (SGS) TS to be consistent with the improved Standard Technical Specifications for Westinghouse plants, NUREG–1431 Revision 2, 4/30/01. The new TS are not an accident initiator. Specifying a minimum boron concentration in a new TS and relocating fuel assembly storage requirements in a new TS are conservative approaches to operational control.

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

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

Response: No.

Criticality accidents in the spent fuel pool have been analyzed in the previous criticality safety analyses documented in PSEG letter NLR–N93058 dated April 28, 1993 transmitting License Change Request (LCR) 93–02 and Attachment D, The Licensing Report for Spent Fuel Storage Capacity Expansion, Public Service Electric and Gas Company, Salem Generating Stations 1 & 2, USNRC [U.S. Nuclear Regulatory Commission] Docket Nos[.] 50–272 & 50–311, prepared by Holtec International. This is the basis for the present TS. The addition of a Limiting Condition for Operation (LCO) for boron concentration does not alter the assumptions or the results of the existing spent fuel criticality analyses or accident analyses described in the Salem Updated Final Safety Analysis Report. The addition of TS which provide for TS control where previous administrative controls had been in place and relocation of material within existing TS does not alter the results of criticality safety analyses.

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

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

Response: No.

The TS changes proposed and the resulting spent fuel storage operation limits will continue to provide adequate safety margin to ensure that the stored fuel assembly array will remain subcritical. Those limits are based on a plant specific criticality analysis and are unchanged by this application. The addition of TS which provides for TS control where previous administrative controls had been in place and relocation of material within existing TS continue to establish conservative operational control.

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: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, PO Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James Andersen, Acting.

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

Date of amendment request: October 25, 2002.

Description of amendment request: The proposed amendment would revise the numerical value of the Safety Limit Minimum Critical Power Ratio (SLMCPR) in Technical Specification (TS) 2.1.1.2 to incorporate the results of the cycle-specific core reload analysis for Browns Ferry Unit 2 Cycle 13 operation.

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

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

Response: No.

The proposed amendment establishes a revised SLMCPR value for two recirculation loop operation. The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The proposed SLMCPR preserves the existing margin to transition boiling and the probability of fuel damage is not increased. Since the change does not require any physical plant modifications or physically affect any plant components, no individual precursors of an accident are affected and the probability of an evaluated accident is not increased by revising the SLMCPR value.

The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. The revised SLMCPR has been determined using NRC-approved methods and procedures. The basis of the MCPR Safety Limit is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. These calculations do not change the method of operating the plant and have no effect on the consequences of an evaluated accident. Therefore, the proposed TS change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed license amendment involves a revision of the SLMCPR for two recirculation loop operation based on the results of an analysis of the Cycle 13 core. Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in the allowable methods of operating the facility. This proposed license amendment does not involve any modifications of the plant configuration or changes in the allowable methods of operation. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The margin of safety as defined in the TS bases will remain the same. The new SLMCPR was calculated using NRC-approved methods and procedures, which are in accordance with the current fuel design and licensing criteria. The SLMCPR remains high enough to ensure that greater than 99.9 percent of all fuel rods in the core are expected to avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity. Therefore, the proposed TS change does not involve a 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: 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. 50–327, Sequoyah Nuclear Plant (SQN), Unit 2, Hamilton County, Tennessee

Date of amendment request: November 15, 2002.

Description of amendment request: The proposed one-time condition would establish special provisions and requirements for safe operation of Unit 2 while heavy load lifts are performed on Unit 1. The provisions for heavy load lifts are described in Topical Report 24370–TR–C–002 that was previously submitted on April 15, 2002, for NRC review and approval. The topical report contains prerequisite actions for heavy load movement, active monitoring during heavy load movement, and compensatory measures in response to the unlikely event of a heavy load drop. This submittal withdraws an amendment request dated July 10, 2002.

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

No changes in event classification as discussed in SQN Updated Final Safety Analysis Chapter 15 will occur due to the proposed license amendment. The one-time provision ensures that the SQN ERCW [essential raw cooling water] system remains functional for continued safe operation of Unit 2 during heavy load lifts performed on Unit 1 during SGR (steam generator replacement) replacement [sic] activities.

Accordingly, the proposed modification to SQN Unit 2 operating license and the implementation of compensatory measures for a postulated load drop will not significantly increase 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.

The possibility of a new or different accident scenario occurring as a result of activities conducted during the SQN Unit 1 SGR project are [sic] not created. Three postulated scenarios related to heavy load handling during the SGR project were examined for their potential to represent a new or different kind of accident from those previously evaluated: (1) A breach of the old steam generator (OSG), resulting in the release of contained radioactive material, (2) flooding in the Auxiliary Building caused by the failure of piping in the ERCW tunnel, and (3) loss of ERCW to support safe shutdown of the operating unit.

Failure of an OSG that results in a breach of the primary side of the steam generator (SG) could potentially result in a release of a contained source outside containment. The consequences of this event, both offsite and in the control room, were examined and found to be within the consequences of the failure of other contained sources outside containment at the SQN site (*i.e.*, within the SQN design basis).

With regard to flooding of the Auxiliary Building from a heavy load drop, the protective measure taken prior to the lifting of heavy loads include installation of a wall in the ERCW tunnel near the Auxiliary Building interface. The wall provides protection against a postulated flood of the ERCW tunnel and protects against flooding of the Auxiliary Building beyond those events previously evaluated.

With regard to the potential for a heavy load drop causing the loss of ERCW cooling water to the operating unit (*i.e.*, Unit 2), TVA is implementing provisions to preclude a load drop. A heavy load drop is considered an unlikely accident for the following reasons:

The lifting equipment was specifically designed and chosen for the subject heavy lifts,

—Crane operators will be specially trained in the operation of the lift equipment and in the SQN site conditions,

—Qualifying analyses and administrative controls will be used to protect the lifts from the effects of external events,

The areas over which a load drop could cause loss of ERCW are a small part of the total travel path of the loads.

In addition, protection against the potential for a loss of ERCW is established prior to any heavy load lifts. Compensatory measures ensure the ERCW system is isolated should a pipe break occur, and that ERCW flow is redirected to equipment essential for safe shutdown capability of Unit 2.

Accordingly, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

The proposed change to the Unit 2 operating license supports safe operation and safe shutdown capabilities of Unit 2 during replacement of the Unit 1 SGs. These measures do not result in changes in the design basis for plant structures, systems, and components (SSCs). Consequently, the proposed change will not affect any margins of safety for plant SSCs.

Accordingly, a significant reduction in the margin of safety is not created by the proposed change.

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

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

NRC 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, 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 email to *pdr@nrc.gov.*

Carolina Power & Light Company, Docket Nos. 50–324, Brunswick Steam Electric Plant, Unit 2, Brunswick County, North Carolina

Date of amendment request: November 26, 2001, as supplemented January 31, February 5, February 11, and October 8, 2002.

Description of amendment request: The amendment revises the Improved Technical Specification 5.5.12 to allow a one-time interval increase for the Type A Integrated Leakage Rate Test for no more than 2 years, 2 months.

Date of issuance: November 21, 2002.

Effective date: November 21, 2002.

Amendment No.: 250.

Facility Operating License No. DPR–62: The amendment changes the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 8, 2002 (67 FR 926). The January 31 and February 5, 2002, supplements contained clarifying information only, and did not change the initial no significant hazards consideration determination or expand the scope of the initial **Federal Register** notice. The February 11 and October 8, 2002, supplements revised the original requests but the initial no significant hazards determination bounded the revised request.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 21, 2002.

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut, Inc., Docket No. 50–423, Millstone Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: July 19, 2002, as supplemented September 6, 2002.

Brief description of amendment: The amendment revised Technical Specification (TS) Surveillance Requirement (SR) 4.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period is extended from the current limit of ''* * * up to 24 hours'' to '' * * * up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater.'' In addition, the following requirement is added to SR 4.0.3: ''A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed.'' The amendment also made administrative changes to SRs 4.0.1 and 4.0.3 to be consistent with NUREG–1431, Revision 2, ''Standard Technical Specifications—Westinghouse Plants.''

Date of issuance: November 15, 2002.

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

Amendment No.: 213.

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

*Date of initial notice in **Federal Register:*** September 4, 2002 (67 FR 56604).

The September 6, 2002, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 15, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 16, 2002, as supplemented by letter dated September 4, 2002.

Brief description of amendment: The amendment changes the technical specifications (TS) to revise the specified minimum emergency diesel generator (DG) steady state output voltage from 3740 volts to 3910 volts.

Date of issuance: November 14, 2002.

Effective date: November 14, 2002, to be implemented within 30 days from the date of issuance.

Amendment No.: 181.

Facility Operating License No. NPF–21: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 20, 2002 (67 FR 53985).

The September 4, 2002, supplemental letter provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 14, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 30, 2002, as supplemented on September 13 and November 6 and 20, 2002.

Brief description of amendment: The amendment revises the Facility Operating License and the Technical Specifications to increase the licensed core thermal power level to 3067.4 megawatts (MWt), which is a 1.4% increase above the currently authorized power level of 3025 MWt. The power uprate is based on the improvement in the core power uncertainty allowance originally required for the emergency core cooling system (ECCS) evaluations performed in accordance with Appendix K, ''ECCS Evaluation Models,'' to part 50 of Title 10 of the CFR. Specifically, the reduced uncertainty is obtained by using a more accurate measurement of feedwater flow.

Date of issuance: November 26, 2002.

Effective date: November 26, 2002.

Amendment No.: 213.

Facility Operating License No. DPR–64: Amendment revised the Technical Specifications and License.

*Date of initial notice in **Federal Register:*** July 9, 2002 (67 FR 45565).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 26, 2002.

The September 13, November 6, and November 20, 2002, letters provided clarifying information that did not enlarge the scope of the amendment request or change the initial proposed no significant hazards consideration determination.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3, York County, Pennsylvania

Date of application for amendment: May 24, 2002, as supplemented by letters dated June 27, September 11, September 24, and October 16, 2002.

Brief description of amendments: The amendments increase the licensed power level by approximately 1.62% from 3458 megawatts thermal (MWt) to 3514 MWt. These changes are based on increased feedwater flow measurement accuracy achieved by utilizing high accuracy ultrasonic flow measurement instrumentation.

Date of issuance: November 22, 2002.

Effective date: For Peach Bottom Atomic Power Station, Unit 2, as of the date of issuance and shall be implemented within 60 days of issuance. For Peach Bottom Atomic Power Station, Unit 3, as of its date of issuance, and shall be implemented upon startup following the Unit 3 14th Refueling Outage, currently scheduled for fall 2003.

Amendment No: 247 and 250.

Facility Operating License Nos. DPR–44 and DPR–56: The amendment revises the Technical Specifications and License.

*Date of initial notice in **Federal Register:*** July 9, 2002 (67 FR 45568).

The June 27, September 11, September 24, and October 16, 2002, supplemental letters 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 November 22, 2002.

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: June 12, 2000, as supplemented by letters dated November 7, 2000, June 19 and August 17, 2001, January 15, June 5, and September 20, 2002.

Brief description of amendments: The amendments replace the current accident source term used in design-basis radiological analyses for control room habitability with an alternative source term (AST) pursuant to Title 10 of the CFR part 50.67, ''Accident Source Term.'' The licensee for D.C. Cook, Units 1 and 2, Indiana Michigan Power Company has requested a selective implementation of the AST limited to control room habitability assessments. The licensee has elected to use the AST and its associated acceptance criteria in preparing a revised control room dose analysis to show compliance with 10 CFR Part 50, Appendix A Criterion 19 ''Control Room.''

In addition, the proposed amendments revise the technical specifications (TSs) to change the standard by which charcoal used in engineered safeguard features systems is tested. The proposed changes to the TSs are made in accordance with Generic Letter 99–02, ''Laboratory Testing of Nuclear-grade Activated Charcoal.'' The amendments also revise the format of the TS pages to adopt the format of Technical Specification Task Force (TSTF) Document TSTF–287 ''Ventilation System Envelope Outage Time.''

Date of issuance: November 14, 2002.

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

Amendment Nos.: 271 and 252.

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

*Date of initial notice in **Federal Register**:* August 23, 2000 (65 FR 51356).

The supplemental letters provided by the licensee contained clarifying information and did not change the initial no significant hazards consideration 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 November 14, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 14, 2002.

Brief description of amendments: The amendments would revise Unit 2 technical specification (TS) 3.4.2, ''Safety Valves—Shutdown,'' and TS 3.4.3, ''Safety Valves—Operating,'' to increase the allowable as-found setpoint tolerance for the Unit 2 pressurizer code safety valves from plus or minus (±) 1 percent (%) to ±3%. In addition, the amendment would add an allowable ±1% as-left setpoint tolerance for the pressurizer code safety valves to Unit 1 and Unit 2 TS 3.4.2 and TS 3.4.3.

Date of issuance: November 26, 2002.

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

Amendment Nos.: 272 and 253.

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

*Date of initial notice in **Federal Register**:* April 2, 2002 (67 FR 15624).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 26, 2002.

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: July 22, 2002.

Brief description of amendment: The amendment removes from Technical Specification (TS) 2.10.4(4)a and b, ''Azimuthal Power Tilt (T),'' the reference to a specific computer program for monitoring core radial peaking factors when a core power tilt is present. Instead, the functional requirement is specified. This change clarifies the requirements for core tilt monitoring associated with a computer system upgrade and changes in computer programs. Also, a clarification is made in the Bases section for TS 2.10.4 regarding the application of TS 2.10.4(1)(b) when the plant computer incore detector alarms for monitoring core linear heat rate become inoperable.

Date of issuance: October 29, 2002.

Effective date: October 29, 2002, and shall be implemented within 120 days from the date of issuance.

Amendment No.: 211.

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

*Date of initial notice in **Federal Register**:* September 3, 2002 (67 FR 56326).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 29, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 11, 2002.

Brief description of amendments: The amendments revise Technical Specification 3.6.4, ''Containment Pressure,'' to reduce the maximum allowable pressure from 3 pounds per square inch gauge (psig) to 2 psig.

Date of issuance: November 26, 2002.

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

Amendment Nos.: 206 and 211.

Facility Operating License Nos. DPR–24 and DPR–27: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* March 19, 2002 (67 FR 12605).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 26, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 28, 2000, as supplemented by letters dated March 29, 2001; October 31, 2001; December 21, 2001; and October 18, 2002.

Brief description of amendment: The amendment replaces the current technical specifications with a set of permanently defueled technical specifications (PDTS) to reflect the permanently defueled condition of the plant.

Date of issuance: November 18, 2002.

Effective date: November 18, 2002, and shall be implemented within 60 days of issuance, including the incorporation of the revised Quality Assurance Program description that contains the relocated administrative control requirements as described in the licensee's March 29, October 31, and December 21, 2001 letters.

Amendment No.: 34.

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

*Date of initial notice in **Federal Register**:* December 26, 2001 (66 FR 66471).

The December 21, 2001, and October 18, 2002, supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff original no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 18, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 20, 2002.

Brief description of amendments: The amendments revised TS Table 3.3.6.1–1, ''Primary Containment Isolation Instrumentation,'' Functional Unit 5.a, Reactor Water Cleanup System Isolation, Main Steam Valve Vault Area Temperature—High, to extend the frequency of the channel calibration surveillance requirement from 122 days to 24 months, and revised applicable Bases.

Date of issuance: November 26, 2002.

Effective date: As of date of issuance and shall be implemented within 60 days from the completion of Browns Ferry Units 2 and 3 refueling outages currently scheduled for early 2003, and the spring of 2004, respectively.

Amendment Nos.: 277 and 236.

Facility Operating License Nos. DPR–52 and DPR–68: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register****:* October 15, 2002 (67 FR 63698).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 26, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 14, 2002.

Brief description of amendment: The amendment reduced the steady-state specific activity of the primary coolant. The amendment also changes the allowable value for the main control room air intake radiation monitor made necessary by reducing the specific activity.

Date of issuance: November 18, 2002.

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

Amendment No.: 41.

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

Date of initial notice in ***Federal Register:*** April 2, 2002 (67 FR 15629).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 18, 2002.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 2nd day of December 2002.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. 02–30921 Filed 12–9–02; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

SES Performance Review Board

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: Notice is hereby given of the appointment of members of the OPM Performance Review Board.

FOR FURTHER INFORMATION CONTACT: Teresa Floyd, Office of Human Resources and EEO, Office of Personnel Management, 1900 E Street, NW., Washington, DC 20415, (202) 606–2309.

SUPPLEMENTARY INFORMATION: Section 4314(c) (1) through (5) of title 5, U.S.C., requires each agency to establish, in accordance with regulations prescribed by the Office of Personnel Management, one or more SES performance review boards. The board reviews and evaluates the initial appraisal of a senior executive's performance by the supervisor, and considers recommendations to the appointing authority regarding the performance of the senior executive.

Office of Personnel Management.

Kay Coles James,

Director.

The following have been designated as regular members of the Performance Review Board of the Office of Personnel Management:

Paul T. Conway, Chief of Staff—Chair.
Kathy L. Dillaman, Acting Director, Investigations Service.
William E. Flynn, Senior Policy Advisor to the Director.
John C. Gartland, Director, Office of Congressional Relations.
Doris L. Hausser, Acting Director, Workforce Compensation and Performance Service.
Teresa M. Jenkins, Director, Office of Workforce Relations.
Gail Lovelace, Chief People Officer, General Services Administration.
Mark A. Robbins, General Counsel.

[FR Doc. 02–31085 Filed 12–9–02; 8:45 am]

BILLING CODE 6325–45–P

SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–46942; File No. SR–NASD–99–60]

Self-Regulatory Organizations; Notice of Filing of Amendment Nos. 3 and 4 to a Proposed Rule Change by the National Association of Securities Dealers, Inc. Regarding Restrictions on the Purchase and Sale of Initial Public Offerings of Equity Securities

December 4, 2002.

On October 15, 1999, the National Association of Securities Dealers, Inc. (''NASD'') filed with the Securities and Exchange Commission (''Commission'' or ''SEC''), pursuant to Section 19(b)(1) of the Securities Exchange Act of 1934 (''Act'') [1] and Rule 19b–4 thereunder,[2] a proposed rule change that would govern trading in ''hot equity'' offerings. The proposed rule, NASD Rule 2790, would revise and replace NASD IM–2110–1, known as the Free-Riding and Withholding Interpretation. On December 21, 1999, the NASD submitted Amendment No. 1 to the proposed rule change.[3] The proposed rule change and Amendment No. 1 were published for comment in the **Federal Register** on January 18, 2000.[4] On October 11, 2000, the NASD submitted Amendment No. 2 to the proposal [5] which, among other things, changed the subject of the proposed rule from ''hot issues'' to ''new issues.'' Amendment No. 2 was published for comment in the **Federal Register** on December 6, 2000.[6] The NASD submitted Amendment No. 3 to the proposal on March 20, 2001,[7] and Amendment No. 4 to the proposal on

[1] 15 U.S.C. 78s(b)(1).

[2] 17 CFR 240.19b-4.

[3] *See* Letter from Gary L. Goldsholle, NASD, to Katherine A. England, Division of Market Regulation, SEC, dated December 20, 1999 (''Amendment No. 1''). In Amendment No. 1, the NASD made certain technical amendments to the proposed rule change.

[4] Securities Exchange Act Release No. 42325 (January 10, 2000), 65 FR 2656 (''Original Notice'').

[5] *See* Letter from Alden S. Adkins, NASD, to Katherine A. England, Division of Market Regulation, SEC, dated October 10, 2000 (''Amendment No. 2'').

[6] Securities Exchange Act Release No. 43627 (November 28, 2000), 65 FR 76316 (''Amendment No. 2 Notice'').

[7] *See* Letter from Patrice M. Gliniecki, NASD, to Katherine A. England, Division of Market Regulation, SEC, dated March 20, 2001 (''Amendment No. 3'').