those procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting. Persons desiring to make oral statements should notify the Associate Director for Technical Support named below five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Associate Director prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Associate Director if such rescheduling would result in major inconvenience.

In accordance with subsection 10(d) Pub. L. 92–463, I have determined that it is necessary to close a portion of this meeting noted above to discuss and protect information classified as national security information pursuant to 5 U.S.C. 552b(c)(1).

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, as well as the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Dr. Sher Bahadur, Associate Director for Technical Support (301) 415–0138), between 7:30 a.m. and 4:15 p.m., ET.

ACRS meeting agenda, meeting transcripts, and letter reports are available through the NRC Public Document Room at *pdr@nrc.gov*, or by calling the PDR at 1–800–397–4209, or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS) which is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* or *http://www.nrc.gov/reading-rm/doc-collections/* (ACRS & ACNW Mtg schedules/agendas).

Videoteleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician (301) 415–8066), between 7:30 a.m. and 3:45 p.m., ET, at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment and facilities that they use to establish the videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

Dated: May 20, 2003.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 03–13142 Filed 5–23–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Meeting of the Subcommittee on Plant License Renewal; Notice of Meeting

The ACRS Subcommittee on Plant License Renewal will hold a meeting on June 11, 2003, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows: *Wednesday, June 11, 2003—8:30 a.m. until the conclusion of business.*

The purpose of this meeting is to review the license renewal application for the Fort Calhoun Station Unit 1 and the NRC staff's initial Safety Evaluation Report. The Subcommittee will hear presentations by and hold discussions with representatives of the NRC staff, the Omaha Public Power District, and other interested persons regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Ralph Caruso (telephone 301/415–8065), five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 4:15 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes to the agenda.

Dated: May 19, 2003.

Sher Bahadur,

Associate Director, for Technical Support, ACRS/ACNW.

[FR Doc. 03–13143 Filed 5–23–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from, May 2, 2003, through May 15, 2003. The last biweekly notice was published on May 13, 2003 (68 FR 25648).

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

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

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By June 26, 2003, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff, or may be delivered to the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. Because of continuing disruptions in delivery of mail to United States Government offices, it is requested that petitions for leave to intervene and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301–415–1101 or by e-mail to *hearingdocket@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and because of continuing disruptions in delivery of mail to United States Government offices, it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

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

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

AmerGen Energy Company, LLC, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: April 21, 2003.

Description of amendment request: The licensee proposed to revise Sections 3.7 and 4.7, ''Auxiliary Electrical Power,'' of the Technical Specifications (TSs) to make them generally consistent with Nuclear Regulatory Commission (NRC) guidance set forth in NUREG–1433, ''Standard Technical Specifications, General Electric Plants, BWR [Boiling Water Reactor]/4,'' Revision 2, and with the NRC-approved industry guidance identified as Technical Specification Task Force (TSTF) traveler TSTF–360, Revision 1. The amendment would also add a new Section 6.8.5, ''Station Battery Monitoring and Maintenance Program.'' The resulting Sections 3.7, 4.7, and 6.8.5 will be explicitly applicable to station batteries B and C, both safety-related subsystems, and their associated battery chargers. The proposed amendment would revise requirements concerning surveillance, monitoring, and maintenance of the subject batteries and chargers.

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) and performed its own. 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 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). Consequently, all SSCs will continue to perform their design functions with no decrease in their capabilities to mitigate the consequences of postulated accidents. Station battery surveillance, monitoring, and maintenance were not previously factored into the probability of accidents, nor were they factored into scenarios of previously analyzed accidents. Consequently, the proposed revision to Sections 3.7, 4.7, and 6.8.5 of the TSs 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: Kevin P. Gallen, Morgan, Lewis & Bockius, LLP, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Richard J. Laufer.

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

Date of amendments request: April 15, 2003.

Description of amendments request: The amendments would revise Sections 2.2, ''SL [Safety Limits] Violations,'' for reporting such violations to positions in the plant organization; 5.2.1, ''Onsite and Offsite Organization,'' for the position responsible for overall safe plant operation; and 5.5.1, ''Offsite Dose Calculation Manual (ODCM),'' for the position that approves changes to the ODCM, of the Technical Specifications (TSs). The revisions would account for the elimination of the positions of Vice President, Nuclear Production, and Director, Site Chemistry, and the assignment of the responsibilities of these positions in the above TS sections to other positions in the plant organization. Also, there would be the format change of adding the title of Section 2.2 near the top of TS page 2.0–2.

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

These changes involve minor changes in the organization of PVNGS. It is expected that the organizational changes will have a positive effect on the conduct of plant operations and safety-related work. Functions which are necessary to operate the facility safely and in accordance with the operating licenses, remain in the re-aligned organization and will not affect the safe operation of the plant and continue to ensure proper control of administrative activities. The Quality Assurance (QA) organization reporting structure has not been affected by these changes allowing the QA organization to maintain the required authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions. The proposed changes will not affect the operation of structures, systems, [or] components, and will not reduce programmatic controls such that plant safety would be affected. (The changes in the plant organization are also not initiators of an accident.) Therefore, the proposed changes do 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 changes will not affect the operation of structures, systems, [or] components, and will not reduce programmatic controls such that plant safety would be affected. The changes in the organization will continue to provide necessary oversight and control of administrative processes. [The changes in the plant organization are also not initiators of an accident.] Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

These changes are administrative and will not diminish any organizational or administrative controls currently in place. The proposed changes will not affect the operation of structures, systems, [or] components, and will not reduce programmatic controls such that plant safety would be affected. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, APS concludes that the activities associated with the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92 ''Issuance of Amendment,'' (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: Kenneth C. Manne, Senior Attorney, Arizona Public Service Company, PO Box 52034, Mail Station 7636, Phoenix, Arizona 85072–2024.

NRC Section Chief: Stephen Dembek.

Calvert Cliffs Nuclear Power Plant, Inc., Docket No. 50–317, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Calvert County, Maryland

Date of amendment request: May 1, 2003.

Description of amendment request: The proposed amendment would increase the maximum enrichment limit of the fuel assemblies that can be stored in the Unit 1 spent fuel pool by taking credit for soluble boron in maintaining acceptable margins of subcriticality.

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

The proposed change will increase the maximum enrichment limit of the fuel assemblies that can be stored in the Unit 1 spent fuel pool (SFP) by taking credit for soluble boron in maintaining acceptable margins of subcriticality. The proposed change will modify Technical Specification 4.3.1 ''Criticality'' and add Technical Specification 3.7.16 ''Spent Fuel Pool Boron Concentration.'' The postulated accidents for the SFP are basically four types: (1) dropped fuel assembly on top of the storage rack, (2) a misloading accident, (3) an abnormal location of a fuel assembly, and (4) loss-of-normal cooling to the SFP.

There is no increase in the probability of a fuel assembly drop accident in the SFP when considering the higher enriched fuel or the presence of soluble boron in the SFP water. Dropping a fuel assembly on top of the SFP storage racks is not credible at Calvert Cliffs due to the design of the spent fuel handling machine and due to the height of the SFP storage racks. The handling of the fuel assemblies has always been performed in borated water and will not change as a result of crediting soluble boron in the SFP criticality analysis. The proposed change does not change the general design and characteristics of the fuel assemblies. Therefore, the proposed change does not increase the probability of a fuel assembly drop accident.

There is no increase in the probability of the accidental misloading of irradiated fuel assemblies into the SFP storage racks when considering the higher enriched fuel or the presence of soluble boron in the SFP water for criticality control. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures.

Due to the design of the SFP storage racks, an abnormal placement of a fuel assembly into the SFP storage racks is not possible. Also, the design of the SFP prevents an inadvertent placement of a fuel assembly between the outer most storage cell and the pool wall. The proposed change does not make any change to the design of SFP. Therefore, there is no increase in the probability of abnormal placement of a fuel assembly into the SFP storage racks.

The proposed change will not result in any changes to the SFP cooling system, and the fuel assembly design and characteristics are not changed by an increase in fuel enrichment. Therefore, there is no increase in the probability of a loss of SFP cooling. Also, since a high concentration of soluble boron has always been maintained in the SFP water, there is no increase in the probability of the loss of normal cooling to the SFP water considering the presence of soluble boron in the pool water for criticality control.

There is no increase in the consequences of an accidental drop or accidental misloading of a maximum enriched fuel assembly into the SFP storage racks, because the criticality analysis demonstrates that the pool will remain subcritical following either event, even if the pool contains a boron concentration less than the proposed Technical Specification limit. The proposed Technical Specification limit will ensure that an adequate SFP boron concentration will be maintained.

There is no increase in the consequences of a loss-of-normal SFP cooling because the Technical Specification boron concentration provides significant negative reactivity. Loss of the SFP water via boiling will not result in a loss of soluble boron, since the soluble boron is not volatile. Therefore, loss of spent fuel pool cooling system without makeup flow is not a mechanism for boron dilution. Even in the unlikely event that soluble boron in the SFP is completely diluted via unborated makeup flow, a pool completely filled with maximum enriched unburned assemblies will remain subcritical by a design margin of k-effective not to exceed 0.986.

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

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

The proposed change will increase the maximum enrichment limit of the fuel assemblies that can be stored in the Unit 1 SFP by taking credit for soluble boron in maintaining acceptable margins of subcriticality. Increasing the maximum enrichment limit does not create a new type of criticality accident.

Soluble boron has been maintained in the SFP water and is currently required by procedures. Therefore, crediting soluble boron in the SFP criticality analysis will have no effect on normal pool operation and maintenance. Crediting soluble boron will only result in increased sampling to verify the boron concentration. This increased sampling will not create the possibility of a new or different kind of accident.

A dilution of the SFP soluble boron has always been a possibility. However, the boron dilution event previously had no consequences, since boron was not previously credited in the accident analysis. The initiating events that were considered for having the potential to cause dilution of the boron in the SFP to a level below that credited in the criticality analyses fall into three categories: dilution by flooding, dilution by loss-of-coolant induced makeup, and dilution by loss-of-cooling system induced makeup. The spent fuel pool dilution analysis demonstrates that a dilution that could increase the rack k-effective greater than 0.95 is not a credible event. It is not credible that dilution could occur for the required length of time without operator notice, since this event would activate the high level alarm and initiate Auxiliary Building flooding. In addition, in excess of 1,043,000 gallons of unborated water must be added to the SFP to reach the minimum soluble boron concentration. This is more water volume than is contained in both pretreated water storage tanks and also more water volume than is contained in the demineralized water storage tank and both condensate storage tanks combined. Even in the unlikely event that soluble boron in the SFP is completely diluted, the SFP will remain subcritical by a design margin of k-effective will not exceed 0.986.

The proposed change will not result in any other change in the plant configuration or equipment design. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The Technical Specification changes proposed by this license amendment request will provide an adequate safety margin to ensure that the stored fuel assembly array of maximum enriched fuel will always remain subcritical. Those limits are based on a plant specific criticality analysis performed for the Calvert Cliffs Unit 1 SFP, that include technically supported margins.

While the criticality analysis utilized credit for soluble boron, the SFP rack k-effective will remain less than 0.986 with no soluble boron with a 95 percent probability at a 95 percent confidence level. This substantial reduction in the SFP soluble boron concentration was evaluated and shown not to be credible. Soluble boron is used to provide subcritical margin such that the spent fuel pool k-effective is maintained less than or equal to 0.95. Since k-effective is less than or equal to 0.95, the current margin of safety is maintained.

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

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

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendments request: April 17, 2003.

Description of amendments request: The proposed amendment would (1) make 19 specific changes to the Technical Specifications actions currently requiring suspension of operations involving positive reactivity additions, and (2) revise various notes precluding reduction in boron concentration. The proposed changes follow the guidance of Technical Specification Task Force (TSTF) Change Traveler 286, Revision 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. Would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The intent of this change is to clarify those Technical Specifications involving positive reactivity additions to the shutdown reactor so that small, controlled, safe insertions of positive reactivity will be allowed where they are now categorically prohibited, posing operational difficulties. These controlled activities could result in a slight change in the probability of an event occurring as Reactor Coolant System (RCS) manipulations that are currently prohibited would now be allowed. However, RCS manipulations are rigidly controlled to minimize the possibility of a significant reactivity increase. In addition, there is sufficient shutdown margin available in these conditions to allow for these slight reactivity changes without significantly increasing the probability of an accident previously evaluated.

The proposed change does not permit the shutdown margin required by the Technical Specifications to be reduced. While the proposed change will permit changes in the discretionary boron concentration above the technical specification requirements, this excess concentration is not credited in the Updated Final Safety Analysis Report safety analysis. Because the initial conditions assumed in the safety analysis are preserved, no increase in the consequence of an accident previously evaluated would occur. In addition, small temperature changes in the RCS impose reactivity changes by means of the moderator temperature coefficient of reactivity. These small changes are within the required shutdown margin, therefore, there is no increase in the consequence of an accident previously evaluated.

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

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

This proposed amendment allows for minor plant operational adjustments without adversely impacting the safety analysis required shutdown margin. It does not involve any change to plant equipment or the shutdown margin requirements in the Technical Specifications.

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

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

The margin of safety in Modes 3, 4, 5, and 6 is preserved by the calculated shutdown margin which prevents a return to criticality. The proposed change will permit reductions in the discretionary shutdown margin beyond the Technical Specification requirements. However, the shutdown margin required by the Technical Specifications is not changed. The proposed change only affects Reactor Coolant System temperature and boron concentration above the calculated shutdown margin. By not impacting the shutdown margin, the margin of safety is not affected.

Therefore, the proposed change will 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: Jay E. Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: February 13, 2003.

Description of amendment request: The proposed amendment would allow the use of an alternative source term (AST) methodology in accordance with 10 CFR 50.67 based on a reevaluation of the loss-of-coolant accident (LOCA) design-basis accident (DBA). Using an approved AST, the licensee has also proposed changes to increase the allowable secondary containment bypass and main steam isolation valve (MSIV) leakage limits and eliminate the MSIV leakage control system. The licensee also proposed changes to the TS Bases.

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

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

The implementation of AST assumptions has been evaluated in a revision to the analysis of the Loss of Coolant Accident (LOCA) and an update to the analysis of the Fuel Handling Accident (FHA).

Based upon the results of the analyses, it has been demonstrated that, with the requested changes, the dose consequences of these limiting Design Basis Accidents (DBAs) are within the regulatory guidance provided by the NRC for use with the AST. This guidance is presented in 10 CFR 50.67, Regulatory Guide 1.183 [''Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors''], and Standard Review Plan (SRP) Section 15.0.1.

The requirements for MSIV [main steam isolation valve] Leakage Control System operability for eliminating MSIV leakage to the environment are being eliminated. This is acceptable because, with the application of AST, this system is no longer credited in mitigating the consequences of a LOCA or any other DBA.

The proposed changes also increase the limits on maximum allowable leakage from secondary containment bypass and main steam isolation valves, and on unfiltered inleakage into the Control Room. This is acceptable due to the new assumptions used in calculating Control Room and offsite dose following the affected design basis accident using the AST methodology.

The proposed changes do not affect the normal design or operation of the facility before the accident; rather, once the occurrence of an accident has been postulated, the new source term is an input to evaluate the consequence. The radiological consequences of the analyzed DBAs have been evaluated with application of AST assumptions. The results conclude that the radiological consequences remain within applicable regulatory limits. Therefore, 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 application of AST does not affect the design, functional performance or normal operation of the facility. Similarly, it does not affect the design or operation of any component in the facility such that new equipment failure modes are created. Elimination of the MSIV Leakage Control System cannot create a new accident because it is used as a mitigation system to limit MSIV leakage after the accident has occurred. Similarly, the use of Standby Liquid Control System to buffer suppression pool pH to prevent iodine reevolution is another mitigation function credited after the accident has occurred and; therefore, cannot create a new accident.

As such 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.

This proposed license amendment involves changes from the original source term developed in accordance with Technical Information Document (TID) 14844 to a new AST, as described in Regulatory Guide 1.183. The results of the DBA analyses and the requested Technical Specification changes, are subject to revised acceptance criteria. The analyses have been performed using conservative methodologies.

Safety margins and analytical conservatisms have been evaluated and have been found acceptable. The analyzed events have been carefully selected and margin has been retained to ensure that the analysis adequately bounds postulated event scenario. The dose consequences of these limiting events are within the acceptance criteria presented in 10 CFR 50.67, Regulatory Guide 1.183 and SRP Section 15.0.1.

The margin of safety is that provided by meeting the applicable regulatory limits. The effect of relaxation of these design and Technical Specification requirements has been analyzed and doses resulting from the design basis accidents have been found to remain within the regulatory limits. The changes continue to ensure that the doses at the exclusion area and low population zone boundaries, as well as the control room, are within the corresponding regulatory limits.

Therefore, operation of Fermi 2 in accordance with the proposed changes will 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: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: February 13, 2003.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) Section 5.5.10, ''Technical Specification (TS) Bases Control Program,'' to be consistent with changes made to 10 CFR 50.59, which were published in the **Federal Register** on October 4, 1999 (64 FR 53582), and which became effective March 13, 2001.

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 deletes the reference to ''unreviewed safety question'' as defined in 10 CFR 50.59. Deletion of the definition of ''unreviewed safety question'' was approved by the NRC with the revision of 10 CFR 50.59. This change is administrative in nature. Consequently, the probability of an accident previously evaluated is not significantly increased. Changes to the TS Bases are still evaluated in accordance with 10 CFR 50.59. As a result, the probability or consequences of any accident previously evaluated are not significantly affected. There is no increase in the radiological dose at the site boundary for any previously evaluated accident. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (*i.e.*, no new or different types of equipment will be installed) or a change to the methods governing normal plant operation. These changes are considered administrative in nature and do not modify, add, delete, or relocate any technical requirements in the TS. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no direct effect on any of the safety analysis assumptions. Changes to the TS Bases that result in meeting the criteria in paragraph 10 CFR 50.59(c)(2) continue to require NRC approval pursuant to 10 CFR 50.59. This change is administrative in nature based on the revision to 10 CFR 50.59. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Section Chief: L. Raghavan.

Detroit Edison Company (DECo), Docket No. 50–341, Fermi 2, Monroe County, Michigan.

Date of amendment request: March 31, 2003.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Surveillance Requirement (SR) 3.7.3.6 associated with the verification of control room emergency filtration (CREF) system duct work unfiltered inleakage. This amendment request supercedes DECo's previous amendment request dated September 26, 2002, in its entirety. The September 26, 2002, amendment request was previously noticed in the **Federal Register** on November 26, 2002 (67 FR 70765).

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.

This license amendment proposes an alternative test for performing the (Control Room Emergency Filtration) CREF system surveillance associated with measuring the

Control Room Envelope (CRE) unfiltered inleakage. The CREF system provides a configuration for mitigating radiological consequences of accidents; however, it does not involve the initiation of any previously analyzed accident. Similarly, the implementation of compensatory measures to address the failure of the surveillance to meet the design basis unfiltered inleakage limits is required to mitigate the consequences of a radiological release. Therefore, the proposed changes cannot increase the probability of any previously evaluated accident.

The CREF system provides a radiologically controlled environment from which the plant can be safely operated following a radiological accident. Design basis accident analyses conclude that radiological consequences are within the regulatory acceptance criteria. The current TS surveillance (SR 3.7.3.6) measures inleakage from four sections of CREF system duct work outside the CRE that are at negative pressure during accident conditions. The proposed Tracer Gas test provides a measurement of CRE inleakage from all potential sources including the four sections of duct work. Measuring the CRE inleakage using Tracer Gas testing has no effect on the CREF system function. The results of Tracer Gas testing will be evaluated against the assumptions in the approved Alternative Source Term (AST) design basis accident analyses and compensatory measures will be implemented, as necessary, to ensure compliance with 10 CFR 50.67. If compliance with 10 CFR 50.67 cannot be demonstrated or if compensatory measures have been in place for more than 18 months, a conservative plant shutdown will be required to minimize risk. Therefore, the proposed changes do not significantly increase the radiological consequences of any previously evaluated accident.

Based on the above, the proposed changes do not significantly increase 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 accident previously evaluated.

The proposed changes do not alter the design function or operation of the system involved. The CREF system will still provide protection to control room occupants in case of a significant radioactive release. The revised TS surveillance requirements provide an alternative test method that has been widely accepted for the measurement of CRE unfiltered inleakage. The proposed changes do not introduce any new modes of plant or CREF system operation. Therefore, the proposed changes do not create the potential for a new or different kind of accident from any accident previously evaluated.

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

The proposed changes to the Fermi 2 TS surveillance requirements do not affect the radiological release from a design basis accident nor the postulated dose to the control room occupants as a result of the accident. The alternate surveillance test requirements provide an acceptable approach for the measurement of CRE inleakage. Safety margins and analytical conservatisms are included in the analyses to ensure that all postulated event scenarios are bounded. The proposed TS requirements continue to ensure that the radiological consequences at the control room are below the corresponding regulatory guidelines and that compliance with 10 CFR 50.67 and GDC (General Design Criterion)–19 is not affected. Therefore, the proposed changes will 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 amendment request involves no significant hazards consideration.

Attorney for licensee: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: April 7, 2003

Description of amendment request: The proposed amendment would move selected Technical Specification (TS) parameters to the Core Operating Limits Reports (COLR). Specifically, the changes proposed affect TSs 2.2, ''Limiting Safety System Settings, Table 2.2–1;'' 3/4.1.1.1.1, ''Reactivity Control Systems, Boration Control, SHUTDOWN MARGIN—Modes 3, 4, and 5 Loops Filled;'' 3/4.1.1.2, ''Reactivity Control Systems, SHUTDOWN MARGIN—Cold Shutdown—Loops Not Filled;'' 3/4.2.5, ''Power Distribution Limits, DNB Parameters;'' 3/4.3.5, ''Instrumentation, SHUTDOWN MARGIN Monitor;'' 3/4.9.1.1, ''Refueling Operations, Boron Concentration;'' Section 6.9.1.6.a, ''Core Operating Limits Report, Core Operating Limits;'' and Section 6.9.1.6.b, ''Core Operating Limits Report, The Analytical Methods Used to Determine the Core Operating Limits,'' and the corresponding pages and Bases sections.

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

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

The relocation of cycle-specific core operating limits from the technical specifications to the COLR has no influence or impact on the probability or consequences of a Design Basis Accident. Adherence to the COLR and methodologies acceptable for establishing COLR parameters continues to be controlled by Technical Specifications. The proposed amendment still requires exactly the same actions to be taken when or if limits are exceeded. Each accident analysis addressed in the Final Safety Analysis Report (FSAR) will be examined with respect to the changes in cycle-dependent parameters, which are obtained from application of the Nuclear Regulatory Commission (NRC) approved reload design methodologies, to ensure that the transient evaluation of new core designs are bounded by previously accepted analysis. This examination, which will be performed in accordance with the requirements of 10 CFR 50.59, ensures that future designs will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to add new document references to Technical Specification Sections 6.9.1.6.b.16 and 6.9.1.6.b.17 are required to identify the most recent methodology to be used in the Millstone Unit No. 3 Small Break Loss of Coolant Accident (SBLOCA) analysis. Section 6.9.1.6.b.18 is added to describe NRC approved Overpower DT and Overtemperature DT trip function methodology. The use of these methodologies demonstrates that the acceptance criteria for SBLOCA events and Overpower DT and Overtemperature DT are met. This change has no impact on plant equipment operation. Since these changes only affect the method of analysis, they cannot affect the likelihood or consequences of accidents. Therefore, these changes will not increase the probability or consequences of an accident previously evaluated.

Deleting the revision number and the date from the documents contained in Technical Specification Section 6.9.1.6.b.1 and in Technical Specification Sections 6.9.1.6.b.4 through 6.9.1.6.b.10 has no impact on the actual analytical methods used to determine the core operating limits, nor does it affect the likelihood or consequences of accidents. Therefore, this change will not increase the probability or consequences of an accident previously evaluated.

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

As stated earlier, the relocation of the cycle-specific variables to the COLR, adding new document references and deleting the revision number and the date in Technical Specification Section 6.9.1.6.b have no influence or impact, nor does it contribute in any way to the probability or consequences of an accident. No safety related equipment, safety function, or plant operations will be altered as a result of this proposed change. The cycle specific variables are calculated using NRC-approved methods and submitted to the NRC to allow the Staff to continue to trend the values of these limits. The Technical Specifications will continue to require operation within the required core operating limits and appropriate actions will be taken when or if limits are exceeded. Therefore the proposed amendment does not in any way create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes have no impact on plant equipment operation. The proposed changes do not revise any setpoints assumed in the analyses and do not affect the acceptance criteria for SBLOCA analyses. Therefore, the proposed changes will not result in a reduction in a margin of safety.

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

Attorney for licensee: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Waterford, CT 06141–5127.

NRC Section Chief: James W. Clifford.

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: April 10, 2003.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) for the low temperature overpressure protection system. Currently, TS Surveillance Requirement (SR) 3.4.12.5 requires performance of a channel functional test for the power-operated relief valve within 12 hours of decreasing reactor coolant system (RCS) temperature to ≤ 325 °F and every 31 days thereafter. The proposed amendments would revise TS SR 3.4.12.5 to allow the first performance of this surveillance to be within 31 days prior to decreasing RCS temperature to ≤ 325 °F. The proposed amendments also would revise the frequency of the channel calibration in TS SR 3.4.12.7 from 18 months to 6 months.

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:

This is a revision to the Technical Specification (TS) surveillance requirement (SR) for performing the channel functional test (CFT) for the pressurizer [power] [..] operated relief valve (PORV). As such, changing the requirement to perform the first CFT before entering the Low Temperature Overpressure Protection (LTOP) region, rather than after LTOP is required, eliminates removing the PORV from service, in the mode of applicability for the performance of the CFT. This change will decrease the probability of a low temperature overpressurization of the reactor vessel, thereby increasing safety and reducing risk, by maintain(ing) both trains (active and passive) of the LTOP System operable. The change to the frequency for performance of SR 3.4.12.7 is being done to ensure the calibration is performed in a time frame supported by current analysis. The method of test is not changed, only the frequency. This reduction in frequency will not significantly increase the probability or consequences of any accident previously evaluated.

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

This revision will not impact the LTOP evaluation analysis. The timeframe to perform the CFT for the PORV will not change the operation of the PORV or its function during accident conditions. No new or different accidents result from performing the CFT prior to entering LTOP conditions. The revision to SR 3.4.12.7 only changes the frequency of the testing. The method of test is not changed. This change has no effect on the possibility of a new or different kind of accident.

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

The proposed revision will perform the CFT within 31 days prior to entering LTOP conditions, rather than performing the test once LTOP conditions are entered. This allows the CFT, which causes the PORV to be inoperable for a short period of time, to be performed prior to reaching the plant conditions where the PORV is relied upon for LTOP. Performing the CFT within 31 days prior to decreasing RCS temperature to < 325 °[F], rather than after entering these conditions, will not change the margin of safety. Oconee calculations show that a recalibration interval of 6 months for the Reactor Coolant System (RCS) low range pressure instrumentation results in a single-sided 95/95 probability confidence limit of 9.4 psig. This result is bounded by the instrument uncertainty assumed in the LTOP evaluation analysis. The frequency change for SR 3.4.12.7 from 18 months to 6 months does not affect the method of test performance. It only decreases the allowed time between performances to reflect current Oconee analysis. This will not significantly reduce 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: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: July 5, 2002, as supplemented August 13, 2002.

Description of amendment request: The proposed amendment would relocate portions of Technical Specification (TS) 3/4.6.B, ''Primary System Boundary—Coolant Chemistry,'' from the TSs to the Updated Final Safety Analysis Report (UFSAR). The portions of the TS that would be relocated to the UFSAR are the reactor coolant chemistry requirements for conductivity and chloride concentration. Specifically, TSs 3/4.6.B.2, 3/4.6.B.3, and 3.6.B.4 would be relocated to the UFSAR.

Basis for proposed no significant hazards consideration determination: As required by Title 10 of the Code of Federal Regulations (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 is administrative in nature and does not involve the modification of any plant equipment or affect basic plant operation. Conductivity and chloride limits are not assumed to be an initiator of any analyzed event, nor are these limits assumed in the mitigation of consequences of accidents.

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

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

Response: No. The proposed change does not involve any physical alteration of plant equipment and does not change the method by which any safety-related system performs its function. As such, no new or different types of equipment will be installed, and the basic operation of installed equipment is unchanged. The methods governing plant operation and testing remain consistent with current safety analysis assumptions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No. The proposed change represents the relocation of current Technical

Specification requirements to the UFSAR, based on regulatory guidance and previously approved changes for other stations. The proposed change is administrative in nature, does not negate any existing requirement, and does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there are no changes being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. Margins of safety are unaffected by requirements that are retained, but relocated from the Technical Specifications to the UFSAR. 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: J.M. Fulton, Esquire, Assistant General Counsel, Pilgrim Nuclear Power Station, 600 Rocky Hill Road, Plymouth, Massachusetts, 02360–5599.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: May 1, 2003.

Description of amendment request: The proposed amendment would modify the surveillance testing requirements for the containment spray system (CSS) by deleting the requirement to verify the position of valves that are locked, sealed, or otherwise secured in their correct position and replacing the quantitative allowable pump degradation value with a requirement to verify the pumps perform in accordance with the Inservice Testing Program.

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.

Analyzed events are assumed to be initiated by the failure of plant structures, systems, or components. Altering the surveillance requirements for the CSS does not increase the probability that a failure leading to an analyzed event will occur. The CSS components are passive until an actuation signal is generated. This change does not increase the failure probability of the CSS components. Therefore, the probability of occurrence for a previously analyzed accident is not significantly increased.

The CSS is primarily designed to mitigate the consequences of a loss of coolant accident (LOCA) or main steam line break (MSLB) accident. The proposed change does not affect any of the assumptions used in the deterministic LOCA or MSLB analyses. Hence the consequences of accidents previously evaluated do not change.

Therefore, the change associated with modifying the CSS surveillance requirements does not involve an increase in the probability or consequences of any accident previously evaluated.

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

Response: No.

The proposed change does not change the design or configuration of the plant. No new equipment is introduced, nor will any installed equipment be operated in a new or different manner. No changes are proposed to the plant's operating parameters or setpoints at which protective or mitigative actions are initiated. Additionally, no substantive changes are proposed to the procedures which ensure the plant remains within analyzed limits or the procedures relied upon to respond to off-normal events. As such, no new failure modes are being introduced. The proposed change does not alter assumptions made in the safety analysis or licensing basis.

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 associated with modifying the surveillance requirements for the CSS does not affect the limiting conditions for operation used in the deterministic analysis to establish the margin of safety. The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. None of these are adversely impacted by the proposed change. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating a transient event.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: March 11, 2003.

Description of amendment request: The proposed amendment will revise and relocate Surveillance Requirement (SR) 4.0.5 and SR 4.4.9 to the administrative section of the Technical Specifications (TS) under sections 6.5.8 and 6.5.7, respectively. The proposed amendment will also relocate TS 3.4.9, "Reactor Coolant System Structural Integrity" and its Bases to the Waterford Steam Electric Station, Unit 3 (Waterford 3) Technical Requirements Manual (TRM). Additionally, the proposed amendment extends the Waterford 3 flywheel volumetric examination interval to ten years.

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

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

Response: No.

The proposed change to relocate SR 4.0.5 to the administrative section of the TSs, including modifications to the wording to make it consistent with NUREG–1432, will not reduce the current testing and inspection requirements. The performance of a code (American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code) inservice test is not an accident initiator. Verbally issuing relief to the ASME Code by the NRC (Nuclear Regulatory Commission) staff in lieu of written relief does not reduce assurance of the health and safety of the public since the NRC staff still reviews the basis for the relief on its technical merit and the NRC staff still obtains management approval prior to granting the relief.

Inspections of the reactor coolant pump (RCP) flywheels are conducted to detect a flaw in the flywheel prior to it becoming a missile that could damage other portions of the facility. The fracture mechanics analyses conducted as part of NRC approved Topical Report SIR–94–080–A, Rev(ision) 1 shows that a conservatively sized pre-existing crack will not grow to a flaw size necessary to create flywheel missiles within the current or extended life of the facility thus the flywheel will remain intact and perform its function to reduce the rate of decay of coolant flow during a postulated loss of power to the RCP motor. This analysis conservatively assumes minimum material properties, maximum flywheel speed, location of the flaw in the highest stress area, and a number of startup and shutdown cycles higher than expected. Since a conservative flaw in the RCP flywheels will not grow to the allowable flaw size under large break LOCA (loss-of-coolant accident) conditions over the life of the plant, reducing the inspection frequency of the flywheels will not significantly increase the probability or consequences of an accident previously evaluated.

The change to move the surveillance requirements for the RCP flywheels to the programs section of the technical specifications is administrative and has no impact on probability or consequences of an accident.

The change to move TS 3.4.9 to the Waterford 3 TRM will have no adverse effect on plant operation or the availability or operation of any accident mitigation equipment. Changes to the TRM are controlled in accordance with 10 CFR 50.59. Therefore, moving TS 3.4.9 to the Waterford 3 TRM will not adversely impact [as] an accident initiator and can not cause an accident.

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 changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. These changes do not introduce any new failure modes.

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 testing and inspection requirements contained in TS 4.0.5 are governed by 10 CFR 50.55a, ''Codes and Standards.'' The 10 CFR requirements to perform the ASME code testing and inspections will not be reduced by the proposed change. The inspections and tests will continue to be performed as they are currently. The proposed change has no impact on plant equipment operation.

The fracture mechanics analysis conducted in support of extending the RCP flywheel volumetric examination interval from three years to ten years shows that significant conservatism has been used for calculating the allowable flaw size, critical flaw size, and crack growth rate in the RCP flywheels. These include minimum material properties, maximum flywheel accident speed, location of the flaw in the highest stress area, and a number of startup/shutdown cycles eight times greater than expected. Since a postulated flaw in a Waterford 3 flywheel will not grow to the allowable flaw size under normal operating conditions or to the critical flaw size under loss of coolant accident conditions over the life of the plant, reducing the examination requirements for the detection of such cracks over the life of the plant will not involve a significant reduction in the margin of safety. The proposed change has no impact on plant equipment operation.

The change to move the surveillance requirements for the RCP flywheels to the programs section of the technical specifications is administrative and has no impact on plant operation.

Relocation of TS 3.4.9 to the TRM does not imply any reduction in its importance in ensuring that the structural integrity and operational readiness of ASME Code Class 1, 2, and 3 components will be maintained at an acceptable level throughout the life of the plant. The proposed change has no impact on plant operation.

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. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: March 31, 2003.

Description of amendment request: The proposed amendments would revise Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF–11 and NPF–18. Specifically, the proposed change will modify TS 5.7, ''High Radiation Area,'' by incorporating the wording and requirements from NUREG–1434, ''Standard Technical Specifications General Electric Plants, BWR/6,'' Revision 2, dated June 2001.

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 will modify LaSalle County Station (LSCS) TS 5.7, ''High Radiation Area,'' by incorporating into the TS the corresponding wording and requirements from NUREG–1434, ''Standard Technical Specifications General Electric Plants, BWR/6,'' Revision 2, dated June 2001. TS 5.7 establishes the administrative controls on entry into high radiation areas. High radiation area administrative controls are not a precursor to accidents previously evaluated. Thus, the proposed change does not have any effect on the probability of an accident previously evaluated.

The proposed change in administrative controls on entry into a radiation area does not affect the ability of LSCS to successfully respond to previously evaluated accidents and does not affect radiological assumptions used in the evaluations. Thus, the radiological consequences of any accident previously evaluated are not increased.

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

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

The proposed change does not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The proposed change does not introduce any new equipment, modes of system operation or failure mechanisms.

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

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

The proposed change incorporates corresponding wording and requirements from NUREG–1434, into the LSCS TS. The LSCS evaluation of the proposed change concluded the following:

- Both the proposed and current TS 5.7.1 describe the requirements for access into areas that have radiation levels that exceed 100 mrem/hr but are less than or equal to 1000 mrem/hr. The proposed and current TS 5.7.1 are considered to have equivalent level access controls as both contain the need to provide a barricade, conspicuously post the area and issue an RWP to control entrance to the area.
- Proposed TS 5.7.2 and current TS 5.7.4 describe the requirements for access into areas that have radiation levels that exceed 1000 mrem/hr. Proposed TS 5.7.2 and current TS 5.7.4 are considered to have equivalent level access controls as both require these areas to be locked. For those areas where locking is not practical, proposed TS 5.7.2 and current TS 5.7.4 both require the area to be barricaded, conspicuously posted, and have an activated flashing light.
- The proposed change includes the deletion of the use of computer controlled doors in current TS 5.7.2. This description is being removed as computer controlled doors are no longer utilized at LSCS. Rather, manual locking mechanisms are used on doors providing an equivalent level of control.
- Current TS 5.7.4 also discusses ''high-high'' radiation areas. The term ''high-high'' radiation area is a legacy term that is being deleted from the proposed TS. This is an administrative change only to remove an outdated term.

Therefore, LSCS has determined that the proposed change provides an equivalent level of protection as that currently provided.

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

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

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: March 11, 2003.

Description of amendment request: The proposed amendment revises the BVPS–1 and 2 Technical Specifications (TSs) to apply the Westinghouse best-estimate large break loss-of-coolant accident (LOCA) methodology to BVPS–1 and 2. The request is contingent upon Nuclear Regulatory Commission (NRC) approval of the licensee's amendment request for conversion of the BVPS–1 and 2 containments from sub-atmospheric to atmospheric which had previously been requested by letter dated June 5, 2002, and which is currently under NRC staff review.

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

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

No. No physical changes are required as a result of implementing best-estimate loss of cooling accident (LOCA) methodology and associated technical specification changes. The plant conditions assumed in the analysis are bounded by the design conditions for all equipment in the plant. Therefore, there will be no increase in the probability of a loss of cooling accident. The consequences of a LOCA are not being increased, since it is shown that the emergency core cooling system is designed so that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, Paragraph b. No other accident is potentially affected by this change.

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 previously evaluated?

No. There are no physical changes being made to the plants. No new modes of plant operation are being introduced. The parameters assumed in the analysis are within the design limits of the existing plant equipment. All plant systems will perform as designed during the response to a potential accident.

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

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

No. It has been shown that the methodology used in the analysis would more realistically describe the expected behavior of plant systems during a postulated loss of coolant accident. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of loss of coolant accidents with different break sizes, different locations and other variations in properties are analyzed to provide assurance that the most severe postulated loss of coolant accidents are calculated. It has been shown by analysis that there is a high level of probability that all criteria contained in 10 CFR 50.46, Paragraph b are met.

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

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.

Indiana Michigan Power Company, Docket No. 50–316, Donald C. Cook Nuclear Plant, Unit 2, Berrien County, Michigan

Date of amendment request: March 27, 2003.

Description of amendment request: The proposed amendment would amend Unit 2 Technical Specification (TS) Table 3.3–4 and the P–11 setpoint in the Engineered Safety Features Interlock Table as follows:

1. Revise the low pressurizer pressure safety injection (SI) trip setpoint from its current value of greater than or equal to 1900 pounds per square inch gauge (psig), to greater than or equal to 1815 psig.

2. Revise the low pressurizer pressure SI allowable value from greater than or equal to 1890 psig, to greater than or equal to 1805 psig.

3. Revise the P–11 setpoint from its current value of greater than or equal to 2010 psig, to greater than or equal to 1915 psig.

4. Make format changes to the affected TS pages that improve appearance but do not affect any requirements.

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 the consequences of an accident previously evaluated?

Response: No.

[Indiana Michigan Power Company] (I&M) proposes changing the low pressurizer pressure SI trip setpoint, the low pressurizer pressure SI allowable value, the P–11 setpoint, and the format of the associated pages. Neither the change to the low pressurizer pressure SI trip setpoint value and the SI allowable value nor the change to the P–11 setpoint value alter any safety-related components or the means of accomplishing a safety-related function. The change in the values is supported by analyses that demonstrate that applicable acceptance criteria are met when SI is initiated at 1700 psig for a (loss-of-coolant accident) LOCA, a main steam system depressurization event, and a feedwater line break. Because the acceptance criteria are met, there is no significant increase in the consequences of an accident. The format changes are intended to improve readability and appearance, and do not alter any requirements. Thus, neither the probability of an accident nor the consequences of an accident are 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.

I&M proposes changing the low pressurizer pressure SI trip setpoint, the low pressurizer pressure SI allowable value, the P–11 setpoint, and the format of the associated pages. Neither the change to the low pressurizer pressure SI trip setpoint value and the SI allowable value nor the change to the P–11 setpoint value involve changing the design function of any component, and a change in any of the values cannot initiate an accident. The format changes are intended to improve readability and appearance, and do not alter any requirements. Thus, no new accident initiators are introduced, and the possibility of a new or different kind of accident is not created.

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

Response: No.

I&M proposes changing the low pressurizer pressure SI trip setpoint, the low pressurizer pressure SI allowable value, the P–11 setpoint, and the format of the associate pages. The low pressurizer pressure instrument is credited for activating the engineered safety features in the event of a LOCA, a main steam system depressurization event, or a feedwater line break. The low pressurizer pressure SI trip setpoint value and the low pressurizer pressure SI allowable value have been selected to insure that the engineered safety features will be activated as assumed in the safety analysis. Present margins continue to be maintained because the applicable accident analyses criteria continue to be met. No margins of safety are associated with the P–11 setpoint value. The format changes are intended to improve readability and appearance, and do not alter any requirements. Thus, there is no significant reduction in a margin of safety.

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

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: L. Raghavan.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of amendment request: April 24, 2003.

Description of amendment request: Eliminate the requirement for continuous Control Room manning when fuel is stored in the fuel storage pool.

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? Involve a significant increase in the probability or consequence of an accident previously evaluated.

Response: No.

The Defueled Safety Analysis (DSAR) identifies five categories of events: spent fuel criticality accidents, a fuel handling accident, a spent fuel cask drop, spent fuel pool accidents, and a low level waste release incident. There are no active controls in the control room that are required to respond to these events. Actions to mitigate the consequences of these events are taken outside the control room. Emergency response is not adversely affected by this proposed change because the control room is still available to the emergency response team and communications capability and timeliness will not be affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The configuration, operations and accident response of the systems, structures or components that support safe storage of the spent fuel are unchanged by the proposed change to the technical specifications. Current site surveillance requirements ensure frequent and adequate monitoring of system and component functionality. Systems in the Spent Fuel Pool Island will continue to be operated in accordance with current design requirements and no new components or system interactions have been identified. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed change. The proposed technical specification change does not have an adverse affect on any system related to safe storage of spent fuel. Therefore, the proposed technical specifications 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.

All design basis accident acceptance criteria will continue to be met. The margin of safety relative to the cooling of the spent fuel is unaffected by the proposed changes as the spent fuel pool parameters will continue to be monitored at the same frequency as assumed in the accident analysis. The ability of the shift crew to respond to abnormal or accident conditions is unaffected by the proposed change since all controls are located in or near the fuel building and any necessary communications will be handled by the on-shift staff and/or DERO. Therefore, it is concluded that the proposed TS change does not involve a significant reduction in the margin of safety

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

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

Attorney for licensee: Joe Fay, Esquire, Maine Yankee Atomic Power Company, 321 Old Ferry Road, Wiscasset, Maine 04578.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: October 8, 2002.

Description of amendment request: The license amendment request proposes to change the title of (a) Shift Supervisor to Shift Manager, (b) ''Plant Manager'' to ''plant manager,'' (c) ''Vice President—Nuclear'' to ''corporate officer with direct responsibility for the plant,'' (d) ''Radiological Manager'' to ''radiological manager,'' (e) ''Operations Supervisor'' to ''operations supervisor'' and (f) ''Shift Radiological Protection/ Chemistry Technician'' to ''radiation protection technician.'' This proposal includes an Updated Safety Analysis Report (USAR) reference correction resulting from the USAR Rebaseline Project and a correction to the title ''Shift Technical Advisor'' to ''Shift Technical Engineer'' in Technical Specification (TS) Section 5.3.1 so as to be consistent with the title used in TS Section 5.2.2.f. These changes do not eliminate any of the qualifications, responsibilities, or requirements for these 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. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

The title of Shift Manager better conveys the appropriate level of responsibility and authority required of the position. The use of generic personnel titles and correction of the USAR reference are strictly administrative. The qualifications, training, duties and experience required of the individuals remain unchanged. The USAR section to be referenced is physically the same section that was referenced before the USAR renumbering. The requested changes do not involve any change to the design basis of the plant or any structure, system, or component. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

There will be no physical alterations to the plant configuration. No changes in operating mode or limits are proposed. The qualifications, training, duties and experience required of the individuals remain unchanged. The USAR section to be referenced is physically the same section that was referenced before the USAR renumbering. Therefore, these proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change in titles and USAR reference are strictly administrative. The qualifications, training, duties and experience required of the individuals remain unchanged. The USAR section to be referenced is physically the same section that was referenced before the USAR renumbering. The proposed changes do not change any license condition or Technical Specifications safety limit or limiting condition for operation. The changes do not involve modification of the design or operation of any plant system involved with controlling the release of radioactivity to the environment. Therefore, these changes do not involve a significant reduction in a margin of safety.

Based on the above, Nebraska Public Power District concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of ''no significant hazards consideration'' is justified.

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

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

Date of amendment request: May 2, 2003.

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) by replacing the existing Reactor Coolant System (RCS) pressure and temperature (P/T) limit curves for in-service leakage and hydrostatic testing, non-nuclear heatup and cooldown, and criticality (Figure 3.4.9–1, ''Pressure Versus Minimum Temperature Valid to Thirty-two Full Power Years, per Appendix G of 10 CFR 50'') with new, updated P/T limits curves. The replacement curves were generated using an NRC-approved methodology (General Electric Report NEDC–32983P) for determining the neutron fluence on the Reactor Pressure Vessel (RPV) and extends the RPV beltline region to encompass a new limiting component, the recirculation inlet nozzle. The change to Figure 3.4.9–1 would also delete the existing notation that states: ''(Interim Approval Until September 1, 2003).''

The licensee's application for amendment dated May 2, 2003, supersedes and withdraws a previous application dated February 28, 2003, for which the NRC has published a notice of consideration of issuance of amendment, proposed no significant hazards consideration determination, and opportunity for hearing in the **Federal Register** (68 FR 12954, dated March 18, 2003).

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

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

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed by the (American Society of Mechanical Engineers Boiler and Pressure Vessel Code) ASME Code and 10 CFR 50 Appendix G and H and associated guidance documents, such as Regulatory Guide 1.99, Rev. 2, as restrictions on normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause non-ductile failure of the reactor coolant pressure boundary. Thus, they ensure that an accident precursor is not likely. Hence, they are included in the TS as satisfying Criterion 2 of 10 CFR 50.36(c)(2)(ii). The revision of the numerical value of these limits, *i.e.*, new curves, using an NRC-approved methodology, does not change the existing regulatory requirements, upon which the curves are based. Thus, this revision will not increase the probability of any accident previously evaluated.

The proposed change does not alter the design assumptions, conditions, or configuration of the facility or the manner in which the facility is operated or maintained. The proposed changes will not affect any other System, Structure or Component (SSC) designed for the mitigation of previously analyzed events. The proposed change does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of any accident previously evaluated. Thus, the proposed revision of the existing numerical values with the updated figure for the RCS P/T limits, which are based upon an NRC-approved methodology for calculating the neutron fluence on the RPV and new limiting component, will not increase the consequences of any previously evaluated accident.

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

The proposed changes do not involve a physical alteration of the plant (*i.e.*, no new or different type of equipment will be installed) or a change in the processes governing normal plant operation. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice. (Nuclear Management Company, LLC) NMC is only requesting to revise the existing numerical values and update the TS figure for the RCS P/T limits based upon an NRC-approved methodology for calculating the neutron fluence on the RPV, and to reflect a new limiting component. The curves continue to be based upon ASME Code Case N–640, which has been previously approved for use at the [Duane Arnold Energy Center] DAEC.

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

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

The proposed changes do not alter the manner in which Safety Limits, Limiting Safety System Settings or Limiting Conditions for Operation are determined. The setpoints at which protective actions are initiated are not altered by the proposed changes. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. NMC is only requesting to revise the existing numerical values and update the TS figure for the RCS P/T limits based upon an NRC-approved methodology for calculating the neutron fluence, NEDC–32983P-A. The new curves also reflect the addition of a new limiting component, the recirculation inlet nozzle (N2). No other changes to the Limiting Conditions for Operation or any Surveillance Requirements of Technical Specification 3.4.9 are proposed.

10 CFR 50, Appendix G specifies fracture toughness requirements to provide adequate margins of safety during operation over the service lifetime. The values of adjusted reference temperature and upper shelf energy are expected to remain within the limits of Regulatory Guide 1.99, Revision 2 and Appendix G of 10 CFR 50 for at least 32 effective full power years (EFPY) of operation. The safety analysis supporting this change continues to satisfy the ASME Code, including ASME Code Case N–640, and 10 CFR 50, Appendices G and H requirements and associated guidance documents, such as Regulatory Guide 1.99, Rev. 2. Thus, the

proposed changes will not significantly reduce any margin of safety that currently exists.

Based upon the above, NMC has determined that the proposed amendment will not involve a significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based upon 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: Jonathan Rogoff, Esquire, General Counsel, NMC, LLC, 700 First St., Hudson, WI 54016.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: March 13, 2003.

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

The changes are based on NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–413, ''Elimination of Requirements for a Post Accident Sampling System (PASS).'' The NRC staff issued a notice of opportunity for comment in the **Federal Register** on December 27, 2001 (66 FR 66949), on possible amendments concerning TSTF–413, including a model Safety Evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 20, 2002 (67 FR 13027). The licensee affirmed the applicability of the following NSHC determination in its application dated March 13, 2003.

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

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

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

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

The regulatory requirements for the PASS can be eliminated without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post-TMI–2 accident guidance through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of the PASS requirements from TSs (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

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

The elimination of PASS-related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage, and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post-accident confinement of radioisotopes within the containment building.

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

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

The elimination of the PASS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of, and recovery from, reactor accidents, results in a neutral impact to the margin of safety. Methodologies that are not reliant on PASS are designed to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI–2 accident can be adequately met without reliance on a PASS.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: March 21, 2003.

Description of amendment request: The proposed amendments would revise Technical Specifications (TS) Section 5.5.1, ''Offsite Dose Calculation Manual (ODCM),'' to remove reference to the Plant Operations Review Committee review and acceptance of licensee initiated changes to the ODCM.

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 change for TS section 5.5.1.b removes the reference to the Plant Operations Review Committee review and acceptance of licensee initiated changes to the ODCM. This change is an administrative change and does not change plant design or responses.

The proposed change does not involve changing any structure, system, or component, or affect reactor operations. It is not an initiator of an accident and does not change any existing safety analysis previously analyzed in the UFSAR. As such, the proposed change does not involve a significant increase in the probability of an accident previously evaluated. Since the proposed change does not alter the plant design, it does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change for TS section 5.5.1.b removes the reference to the Plant Operations Review Committee review and acceptance of licensee initiated changes to the ODCM. This change is an administrative change and does not change plant design or responses.

The proposed change will not alter any plant design basis or postulated accident. In addition, the proposed change does not impact any plant systems or components.

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

The proposed change for TS section 5.5.1.b removes the reference to the Plant Operations Review Committee review and acceptance of licensee initiated changes to the ODCM. This change is an administrative change and does not change plant design or responses. The proposed change does not impact accident offsite dose, containment pressure or temperature, emergency core cooling system setpoints, reactor protection system settings or any other parameter that could affect 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: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: April 11, 2003 (TS–424).

Description of amendment request: The proposed amendments would reduce the number of Emergency Core Cooling System subsystems that are available in response to certain design basis loss-of-coolant accident (LOCA) scenarios because of TVA's planned restart of Unit 1. The licensee stated that the reduced number has been analyzed and is consistent with the current approved LOCA analysis methodology. The amendments are needed to eliminate the potential for overloading a shutdown board or a diesel generator when both Units 1 and 2 are in-service. The reduction requires a change to the Updated Final Safety Analysis Report.

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

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

No. The proposed amendments revise the actual number of Emergency Core Cooling System (ECCS) subsystems that are available in response to certain design basis Loss of Coolant Accident (LOCA) scenarios. The associated modifications also result in a revision to the number of required channels for the Low Pressure Coolant Injection (LPCI) pump start time delay relay function specified in Technical Specifications. The proposed amendments and Technical Specification changes do not affect any accident precursors. Therefore, the probability of an evaluated accident is not increased.

The reduction in the number of ECCS subsystems that are actually available in response to the bounding LOCA case (A recirculation suction line break with an assumed battery failure) will now be the same as the number of ECCS subsystems evaluated in the current BFN SAFER/GESTR–LOCA analysis. The ECCS performance for the bounding LOCA case has previously been evaluated using the approved SAFER/GESTR–LOCA application methodology and is described in Updated Final Safety Analysis Report (UFSAR) Sections 6.5 and 14.6.3. The revision to the number of required channels for the LPCI pump start time delay relay function does not affect the LOCA analysis. The requirements of 10 CFR 50.46 and Appendix K are met. Therefore, the proposed amendments and Technical Specification changes will not significantly increase the consequences of an accident previously evaluated.

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

No. The proposed amendments revise the number of ECCS subsystems that are actually available in response to certain design basis LOCA scenarios. The proposed Technical Specification changes revise the number of required channels for the LPCI pump start time delay relay function. The proposed amendments and Technical Specification changes do not introduce new equipment, which could create a new or different kind of accident.

No new external threats, release pathways, or equipment failure modes are created. Therefore, the implementation of the proposed amendments and Technical Specification changes will not create a possibility for an accident of a new or different type than those previously evaluated.

3. Do the proposed amendments and Technical Specification changes involve a significant reduction in a margin of safety?

No. The proposed amendments and Technical Specification changes revise the number of ECCS subsystems that are actually available in response to certain design basis LOCA scenarios. The reduction in the number of ECCS subsystems that are actually available in response to the bounding LOCA case (A recirculation suction line break with an assumed battery failure) will now be the same as the number of ECCS subsystems evaluated in the current BFN SAFER/GESTR–LOCA analysis. The ECCS performance for the bounding LOCA case has previously been evaluated using the approved SAFER/GESTR–LOCA application methodology. The revision to the number of required channels for the LPCI pump start time delay relay function does not affect the LOCA analysis. The requirements of 10 CFR 50.46 and Appendix K are met. Therefore, the proposed license amendments and Technical Specification 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: 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 Nos. 50–259, 50–260 and 50–296, Browns Ferry Nuclear Plant (BFN), Units 1, 2 and 3, Limestone County, Alabama

Date of amendment request: April 15, 2003 (TS 409).

Description of amendment request: The proposed amendments are applicable to BFN Units 1, 2, and 3. They would revise Technical Specification (TS) Limiting Condition for Operation 3.7.3, Control Room Emergency Ventilation (CREV) System, and its associated TS Bases to provide specific conditions and required actions that address a degraded main control room boundary. The proposed changes are consistent with the TS Task Force Traveler 287, Revision 5.

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?

2. No. The proposed TS change involves the CREV system, which provides a radiological controlled environment from which the plant can be operated following a design basis accident (DBA). The CREV system is not assumed to be the initiator of any analyzed accident and cannot not [sic] affect the probability of accidents.

The proposed change allows the main control room boundary to be opened intermittently under administrative control, and allows 24 hours to restore the main control room boundary to Operable status before requiring the plant to perform an orderly shutdown. The 24-hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period and TVA's commitment to implement, via administrative controls, appropriate compensatory measures consistent with the intent of 10 CFR part 50, Appendix A, General Design Criteria (GDC) 19. These compensatory measures minimize the consequences of an open main control room boundary and assure that CREV system can continue to perform its function. As such, these changes will not affect the function or operation of any other systems, structures, or components.

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?

No. The proposed change allows the main control room boundary to be opened intermittently under administrative control and allows 24 hours to restore the main control room boundary to Operable status before requiring the plant to perform an orderly shutdown. The 24-hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period and TVA's commitment to implement, via administrative controls, appropriate compensatory measures consistent with the intent of 10 CFR part 50, Appendix A, GDC 19. These compensatory measures minimize the consequences of an open main control room boundary and assure that the CREV system can continue to perform its function. As such, these changes will not affect the function or operation of any other systems, structures, or components.

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?

No. The proposed change allows the main control room boundary to be opened intermittently under administrative control and allows 24 hours to restore the main control room boundary to Operable status before requiring the plant to perform an orderly shutdown. The 24-hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period and TVA's commitment to implement, via administrative controls, appropriate compensatory measures consistent with the intent of 10 CFR part 50, Appendix A, GDC 19. These compensatory measures minimize the consequences of an open main control room boundary and assure that the CREV system can continue to perform its function such that compliance with GDC 19 is maintained.

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

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the 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 Nos. 50–260 and 50–296, Browns Ferry Nuclear Plant (BFN), Units 2 and 3, Limestone County, Alabama

Date of amendment request: April 14, 2003 (TS 425).

Description of amendment request: The proposed amendments would revise two Technical Specification (TS) Limiting Conditions for Operation 3.3.4.1, ''End Of Cycle Recirculation Pump Trip (EOC–RPT) Instrumentation,'' and 3.7.5, ''Main Turbine Bypass System,'' to reference additional core limits adjustment factors for linear heat generation rate for equipment out-of-service conditions. Also, Section b of TS 5.6.5, ''Core Operating Limits Report (COLR),'' would be revised to add references to the Framatome Advanced Nuclear Power (FANP) analytical methods that will be used in the upcoming fuel cycles to determine core operating limits. The above TS changes are needed to support a transition to the use of FANP fuel, and FANP core design and analysis services.

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?

No. Core operating limits are established to support requirements, which in turn ensure that fuel design limits are not exceeded during any conditions of operating transients or accidents. The methods used to determine the limits for each operating cycle are based on methods previously found acceptable by the NRC and are required to be listed in COLR TS Section 5.6.5.b. Accordingly, a change to TS Section 5.6.5.b is requested to include FANP methods in the list of NRC-approved methods applicable to BFN. This TS change also adds provisions that ensure core thermal limits adjustment factors are applied for equipment out-of-service conditions associated with the use of FANP methods for transient analyses. The application of these NRC-approved methods will continue to ensure that acceptable operating limits are established and applied for protection of fuel cladding integrity during transient and accidents.

The requested TS changes do not involve any plant modifications or operational changes that could affect system reliability, performance, or possibility of operator error. The requested changes do not affect any postulated accident precursors, do not affect any accident mitigation systems, and do not introduce any new accident initiation mechanisms.

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?

No. The core operating limits and required limits adjustments for equipment out-of-service conditions will continue to be determined using methodologies that have been approved by the NRC. The limits derived from approved methodologies will provide adequate margins of safety. The proposed changes do not involve any new modes of operation, any changes to setpoints, or any plant modifications, and do not result in any new precursors to an accident.

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.

No. The core operating limits and required limits adjustments for equipment out-of-service will continue to be determined using methodologies that have been approved by the NRC. On this basis, the implementation of the changes does not involve a significant reduction in 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. 50–390, Watts Bar Nuclear Plant (WBN), Unit 1, Rhea County, Tennessee

Date of amendment request: May 1, 2003.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.8.7, ''Inverters—Operating.'' The TS as currently written requires two inverters for each of the four instrument channels. The revision changes the requirement to one inverter for each of the four channels. The amendment is the initial phase of a project that will replace the vital inverters to achieve improvements in the reliability of the 120V AC Vital Instrument Power System.

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?

No. The proposed revisions to WBN's Vital AC Power System do not alter the safety functions of the Vital Inverters or the Unit 1 and Unit 2 120V AC Vital Instrument Power Boards. The initial conditions for the Design Basis Accidents (DBAs) defined in the WBN Updated Final Safety Analysis Report (UFSAR) assume the Engineered Safety Feature (ESF) systems are operable. The vital inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to vital instrumentation so that the fuel, reactor coolant system, and containment design limits are not exceeded. Adding the Unit 2 loads to the Unit 1 inverters does not alter the accident analyses as long as the Unit 1 inverters are capable of handling the additional loads and channel separation is maintained. Design calculations document that the Unit 1 inverters have adequate capacity to support the addition of the Unit 2 loads and no changes are proposed that will impact the separation of the Vital AC Power System. In addition, the redundant capabilities of the Vital AC System as currently described in the UFSAR are not impacted by the proposed amendment.

The inverters and the associated 120V AC Vital Instrument Power Boards are utilized to support instrumentation that monitor critical plant parameters to aid in the detection of accidents and to support the mitigation of accidents, but are not considered to be an initiator of design basis accidents. Based on this and the preceding information, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. When implemented, the proposed TS amendment will allow the Unit 2 Vital Instrument Power Boards to receive their uninterruptible power supply (UPS) power from the Unit 1 inverters instead of the Unit 2 inverters. Calculations have verified that the additional load will not affect the ability of the Unit 1 inverters to perform their intended safety functions. In addition, the inverters and the 120V AC Vital Instrument Power Boards are not considered to be an initiator of a design basis accident. These components provide power to instrumentation that supports the identification and mitigation of accidents as well as system control functions during normal plant operations. The functions of the inverters are not altered by the proposed TS change and will not create the possibility of a new or different accident. Further, the addition of the Unit 2 loads to the Unit 1 inverters is the principal change to the inverter system and this change is bounded by previously evaluated accident analyses. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The plant setpoints and limits that are utilized to ensure safe operation and detect accident conditions are not impacted by the proposed TS amendment. The inverters and the 120V Vital Instrument Power Boards will continue to provide reliable power to safety-related instrumentation for the identification and mitigation of accidents and to support plant operation. Therefore, the margin of safety is not reduced.

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

In conclusion, based on the considerations discussed above, (1) There is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by email to *pdr@nrc.gov.*

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

Date of application for amendment: August 1, 2002, as supplemented on October 18, 2002, and April 17, 2003.

Brief description of amendment: The amendment revises Technical Specification (TS) 3.7.1.1, ''Plant Systems: Turbine Cycle Safety Valves,'' to reflect results of a reanalysis of overpressurization events to allow plant operation, at corresponding reduced power levels, with up to four main steam safety valves in each main steam line inoperable.

Date of issuance: May 7, 2003.

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

Amendment No.: 275.

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

Date of initial notice in ***Federal Register***: September 17, 2002 (67 FR 58638). The supplements dated October 18, 2002, and April 17, 2003, provided additional information which clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 7, 2003.

No significant hazards consideration comments received: No.

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: September 30, 2002, as supplemented by letters dated October 17, 2002 and April 2, 2003.

Brief description of amendments: The amendments revise the Technical Specification to: (1) Modify the Surveillance Requirement to be consistent with the design of the reactor building access openings, (2) modify the frequency of the Surveillance Requirement for visual inspections for the exposed interior and exterior surface of the reactor building, and (3) modify the administrative controls for the containment leakage rate testing program.

Date of issuance: May 8, 2003.

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

Amendment Nos.: 212/193.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register***: November 12, 2002 (67 FR 68733). The supplement dated October 17, 2002, and April 12, 2003, provided clarifying information that did not change the scope of the September 30, 2002, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 8, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 3, 2002, as supplemented by letters dated November 27, 2002, and April 17, 2003.

Brief description of amendment: The amendment allows the addition of depleted uranium to the fuel assembly composition described in Technical Specification (TS) 4.2.1. The amendment also revises TS 5.6.5.b to incorporate the references to the analytical methods to be used to determine core operating limits and removes those references that will no longer be used. The amendment also allows the format for those document references to be revised as described in the staff-approved Industry/TSTF Standard Technical Specification Change Traveler, TSTF–363, ''Revise Topical Report References in ITS 5.6.5, COLR.''

Date of issuance: May 12, 2003.

Effective date: May 12, 2003, and shall be implemented within 30 days from the date of issuance.

Amendment No.: 185.

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

Date of initial notice in ***Federal Register***: October 15, 2002 (67 FR 63693). The November 27, 2002, and April 17, 2003, supplemental letters provided additional clarifying information, did not change the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 12, 2003.

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: June 26, 2002, as supplemented on March 12, 2003.

Brief description of amendment: The amendment revises Technical Specification 5.6.5.b, ''Core Operating Limits Report (COLR),'' to incorporate the reference to Westinghouse Topical Report WCAP–12945–P–A, ''Code Qualification Document for Best Estimate Loss-of-Coolant Analysis,'' dated March 1998. The amendment allows the use of the analytical methodology to determine the core operating limits.

Date of issuance: May 6, 2003.

Effective date: May 6, 2003.

Amendment No.: 217.

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

Date of initial notice in ***Federal Register***: August 6, 2002 (68 FR 50952). The March 12 letter provided clarifying information that did not expand the scope of the ***Federal Register*** notice or change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 6, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 5, 2002, as supplemented on September 27, November 6, November 21, and December 30, 2002; February 4, February 10, March 17, and April 14, 2003.

Brief description of amendment: The amendment increases the licensed power level by 1.5%, from 1998 MWt to 2028 MWt, based on the installation of ultrasonic flow measurement instrumentation resulting in improved feedwater flow measurement accuracy. The amendment changes the Operating License (OL) and Technical Specifications (TSs) to reflect the increase in licensed power level.

Date of issuance: May 9, 2003.

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

Amendment No.: 201.

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

Date of initial notice in ***Federal Register***: September 3, 2002 (67 FR 56322). The supplements dated September 27, November 6, November 21, and December 30, 2002; February 4, February 10, March 17, and April 14, 2003, provided additional information that clarified the application, and did not expand the scope of the application or change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 9, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 27, 2003, as supplemented April 7, 2003.

Brief description of amendments: The amendments revise the Technical Specifications by adding a surveillance requirement to perform a quarterly trip unit calibration of the reactor protection system scram discharge volume water level—high differential pressure switches.

Date of issuance: May 6, 2003.

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

Amendment Nos.: 214/208.

Facility Operating License Nos. DPR–29 and DPR–30: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register***: April 1, 2003 (68 FR 15760). The supplement dated April 7, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 6, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 16, 2002, as supplemented October 17, 2002.

Brief description of amendments: These amendments revised portions of the current Technical Specifications, Section 6.0, ''Administrative Controls,'' to conform with improved Technical Specifications. The conversion is based upon: NUREG–1431, ''Standard Technical Specifications for Westinghouse Plants,'' Revision 2, dated April 2001; ''Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors'' (Final Policy Statement), published on July 22, 1993 (58 FR 39132); and Title 10 of the Code of Federal Regulations (10 CFR), § 50.36, ''Technical Specifications,'' as amended July 19, 1995.

Date of issuance: May 15, 2003.

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

Amendment Nos.: 255 and 136.

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

Date of initial notice in ***Federal Register***: October 29, 2002 (67 FR 66010). The supplement dated October 17, 2002, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 15, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 31, 2002, as supplemented September 11, 2002, January 30, and February 21, 2003.

Brief description of amendment: The amendment revised the Technical Specification Design Feature 5.3.1, Criticality, such that the new fuel (fresh fuel) racks enrichment limit specified in Section 5.3.1.2.a was increased from 4.85 weight percent to a 5.00 weight percent limit.

Date of issuance: May 15, 2003.

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

Amendment No: 135.

Facility Operating License No. NPF–73. Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** September 17, 2002 (67 FR 58645). The September 11, 2002, January 30, and February 21, 2003, letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: March 22, 2002, as supplemented May 13, June 24, July 29, and December 20, 2002.

Description of amendment request: The amendment revises Technical Specifications (TSs) 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 adds a requirement for a TS Bases Control Program to the administrative controls section of TSs and makes 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: May 15, 2003.

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

Amendment No.: 87.

Facility Operating License No. NPF–86: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 21, 2003 (68 FR 2804).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 2003.

No significant hazards consideration comments received: No.

Indiana Michigan Power Company, Docket No. 50–316, Donald C. Cook Nuclear Plant, Unit 2, Berrien County, Michigan

Date of application for amendment: November 15, 2002, as supplemented February 24 and April 25, 2003.

Brief description of amendment: The amendment increases the licensed reactor core power level by 1.66 percent from 3411 megawatts thermal (MWt) to 3468 MWt. The power level increase is considered a measurement uncertainty recapture power uprate.

Date of issuance: May 2, 2003.

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

Amendment No.: 259.

Facility Operating License No. DPR–74: Amendment revises the Operating License and Technical Specifications.

Date of initial notice in ***Federal Register:*** January 21, 2003 (68 FR 2805)

The February 24 and April 25, 2003, supplemental letters provided additional clarifying information that was within the scope of the original application and did not change the Nuclear Regulatory Commission staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 2, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: January 27, 2003.

Brief description of amendment: The amendment makes administrative and editorial changes to the Fort Calhoun Station Technical Specifications 1.3 Basis (1); 2.7 (1)a; 2.7 (1)b; 2.7 (1)d; 2.7 (1)i; 2.7 Basis; 3.0.2; Table 3–5, Item 11; and 3.5(3)ii. The changes are primarily editorial and are typographical changes or corrections.

Date of issuance: May 8, 2003.

Effective date: May 8, 2003, and shall be implemented within 60 days from the date of issuance.

Amendment No.: 218.

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

Date of initial notice in ***Federal Register:*** March 18, 2003 (68 FR 12955).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 8, 2003.

No significant hazards consideration comments received: No.

Southern California Edison Company, et al., Docket No. 50–206, San Onofre Nuclear Generating Station, Unit 1, San Diego County, California

Date of application for amendment: March 11, 2003.

Brief description of amendments: The amendment application requests a revision to the Unit 1 defueled Technical Specifications administrative controls section to propose changes in organizational responsibilities. Specifically, the proposed changes identify that the Vice President, Engineering & Technical Services will be responsible for decommissioning activities. Additionally, the Station Manager will be designated as having approval authority for activities within the Station Manager's organization.

Date of issuance: May 15, 2003.

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

Amendment No.: Unit 1–161.

Facility Operating License No. DPR–13: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 15, 2003 (68 FR 18285).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 19, 2003.

Description of amendment request: The amendments deleted Technical Specification 5.5.3, ''Post Accident Sampling'' and, thereby, eliminated the requirements to have and maintain the post accident sampling system.

Date of issuance: May 9, 2003.

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

Amendment Nos.: 245, 282, 240.

Facility Operating License Nos. DPR–33, DPR–52, and DPR–68: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 18, 2003 (68 FR 12957).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 9, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: July 25, 2002, as supplemented by letters dated February 5 and February 11, 2003.

Brief description of amendments: The amendments change the Comanche Peak Steam Electric Station, Units 1 and 2, Facility Operating Licenses as follows: The license conditions related to Decommissioning Trusts, specified in Sections 2.C.(4)(a), 2.C.(4)(b), 2.C.(4)(d), 2.C(4)(e), and 2.C.(6), are deleted and Section 2.E, which requires reporting any violations of the requirements contained in Section 2.C of the licenses, is deleted. Additionally, Technical Specification Table 5.5–2, ''Steam Generator Tube Inspection,'' Table 5.5–3, ''Steam Generator Repaired Tube Inspection for Unit 1 Only,'' and TS 5.6.10c, ''Steam Generator Tube Inspection Report,'' are revised to delete the requirement to notify the NRC pursuant to § 50.72(b)(2), ''Immediate notification requirements for operating nuclear power reactors,'' of Title 10 of the Code of Federal Regulations (10 CFR) if the steam generator tube inspection results are in a Category C–3 classification.

Date of issuance: May 15, 2003.

Effective date: December 24, 2003, and shall be implemented within 60 days from that date.

Amendment Nos.: 103/103.

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

Date of initial notice in ***Federal Register:*** September 3, 2002 (67 FR 56329).

The February 5, 2003, supplement was the subject of a second no significant hazards consideration determination (68 FR 10282, published March 4, 2003). The February 11, 2003, supplement 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 amendments is contained in a Safety Evaluation dated May 15, 2003.

No significant hazards consideration comments received: No.

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

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

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

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

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

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

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

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By June 26, 2003, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, and electronically on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If there are problems in accessing the document, contact the PDR Reference staff at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. Because of the 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 petition for leave to intervene and request for hearing 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 the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

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

Date of application for amendments: April 25, 2003.

Brief description of amendments: The amendments modify Technical Specification surveillance requirements to provide an alternative means of testing the Unit 2 main steam power operated relief valves, including those that provide the automatic depressurization system and low set relief functions.

Date of issuance: May 8, 2003.

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

Amendment Nos.: 215/209.

Facility Operating License Nos. DPR–29 and DPR–30: The amendments revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. Quad-City Times, dated May 5, 2003. The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received.

The Commission's related evaluation of the amendment, finding of exigent circumstances, state consultation, and final NSHC determination are contained in a Safety Evaluation dated May 8, 2003.

Dated at Rockville, Maryland, this 19th day of May 2003.

For the Nuclear Regulatory Commission.

William H. Ruland,

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

[FR Doc. 03–12973 Filed 5–23–03; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

Issuer Delisting; Notice of Application To Withdraw From Listing and Registration on the American Stock Exchange LLC (Atlantic Premium Brands, Ltd., Common Stock, $.01 par value) File No. 1–13747

May 19, 2003.

Atlantic Premium Brands, Ltd., a Delaware corporation (''Issuer''), has filed an application with the Securities and Exchange Commission (''Commission''), pursuant to section 12(d) of the Securities Exchange Act of 1934 (''Act'')[1] and Rule 12d2–2(d) thereunder,[2] to withdraw its Common Stock, $.01 par value (''Security''), from listing and registration on the American Stock Exchange LLC (''Amex'' or ''Exchange'').

The Issuer stated in its application that it has met the requirements of Amex Rule 18 by complying with all applicable laws in the State of Delaware, in which it is incorporated, and with the Amex's rules governing an issuer's voluntary withdrawal of a security from listing and registration.

The Board of Directors (''Board'') of the Issuer approved a resolution on May 14, 2003 to withdraw the Issuer's Security from listing on the Amex. The Board considered such action to be in the best interest of the Issuer and its stockholders. In addition, the Board states that it took into account alternatives explored by the Issuer, including, without limitation, that: (i) The significant costs associated with maintaining the Issuer's status as a reporting company are expected to increasingly reduce profitability; (ii) the limited volume of trading of the Issuer's Security has resulted in the shares not providing a practical source of capital or liquidity; and (iii) no analysts currently cover the Issuer and its Security. The Issuer states in its application that it is currently seeking to list its Security on the Pink Sheets.

The Issuer's application relates solely to the withdrawal of the Securities from listing on the Amex and from registration under section 12(b) of the

[1] 15 U.S.C. 78*l*(d).

[2] 17 CFR 240.12d2–2(d).