longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to *dkw@nrc.gov*.

Dated: March 14, 2002.

**David Louis Gamberoni,**

*Technical Coordinator, Office of the Secretary.*

[FR Doc. 02–6655 Filed 3–15–02; 10:48 am]

**BILLING CODE 7590–01–M**

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## NUCLEAR REGULATORY COMMISSION

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

### I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from February 22, 2002 through March 7, 2002. The last biweekly notice was published on March 5, 2002 (67 FR 10006).

### 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 NRC's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By April 18, 2002, 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 NRC's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (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, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management 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 PDR Reference staff at 1–800–397–4209, 304–415–4737 or by e-mail to *pdr@nrc.gov*.

*AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania*

*Date of amendment request:* December 19, 2001.

*Description of amendment request:* The proposed amendment provides clarifications and substantive changes to the decay heat removal (DHR) Technical Specifications (TSs). It is intended, in part, to fulfill a commitment made by the licensee to the NRC during a pre-decisional enforcement conference on April 23, 1999. Specifically, the proposed changes would: (1) define and clarify the emergency feedwater (EFW) flowpath redundancy as described in the Bases; (2) provide operability requirements for the redundant steam supply paths to the turbine-driven EFW pump; (3) provide a 72-hour allowed outage time (AOT) with any EFW pump or flowpath inoperable; (4) provide a 24-hour AOT with one steam supply path to the turbine-driven EFW pump and one motor-driven EFW pump inoperable; (5) provide a requirement to initiate action to immediately restore at least 2 EFW pumps and one flowpath to each once-through steam generator (OTSG) if more than one EFW pump or both flowpaths to either OTSG were inoperable; (6) provide a statement suspending actions requiring shutdown or changes in reactor operating conditions until at least 2 EFW pumps and one EFW flowpath to each OTSG are restored to operable status; and (7) revise, relocate and clarify EFW pump and flowpath operability requirements during surveillance testing. Minor administrative and editorial changes are also proposed, including relocation of some requirements for clarity. A note is added to TS 4.9.1.1 and its related Bases to indicate that the surveillance is not applicable to the turbine driven EFW pump until 24 hours after exceeding 750 psig. A change to TS Table 3.5–2 and the Bases for TS 3.5.5, ''Accident Monitoring Instrumentation,'' regarding the description of the pressurizer level instrument channels to reflect the replacement of Bailey transmitters was also included. Unrelated editorial changes to the Table of Contents were also included.

*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. Will the operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. This change incorporates the concept of EFW flowpath redundancy throughout the TS[s], which takes into consideration the redundancy provided by the EFW System modifications made in the mid-1980s after the accident at TMI–2 [Three Mile Island Nuclear Station, Unit 2]. This change incorporates appropriate Limiting Conditions of [for] Operation (LCOs) and required action times and clarifies the design basis of the EFW System technical specification requirements in the LCOs and Surveillance Standards. These changes will not result in any change to the configuration of the EFW System as described in the SAR [Safety Analysis Report] or used in plant specific analyses. The reliability of EFW System components is unaffected. With less than the minimum EFW capability, this change incorporates the STS [standard technical specification] requirement to initiate action immediately to restore EFW components and suspend all actions requiring shutdown or changes in reactor operating conditions. The seriousness of this condition requires that action be started immediately to restore EFW components to operable status prior to power reductions that could result in a plant trip with no safety related means for conducting a cooldown. This change will not significantly affect any accident initiation sequence or the off site dose consequences of accidents that have been analyzed.

The current surveillance standard contains EFW flowpath operability requirements being moved to the Limiting Conditions of [for] Operation (LCO) section in Chapter 3 and combined with the notes to define the EFW System operability requirements for EFW pumps and flowpaths during surveillance testing. The revised specification incorporates consideration of EFW flowpath redundancy consistent with HSPS [heat sink protection system] train operability requirements and continues to require that compensatory measures be implemented to promptly restore components if EFW is needed during surveillance testing when more than one pump or both flowpath[s] to an OTSG are inoperable. The intent of this surveillance standard has been retained, which assures that the minimum number of EFW flowpaths to the OTSGs will be available with minimal operator action. The addition of a note, currently provided in the Standard Technical Specifications which permits a delay in performing the surveillance of the turbine-driven EFW Pump is needed to assure sufficient main steam pressure is available for performance of the test and does not significantly affect the reliability of the pump or the consequences of accidents previously evaluated.

This change provides further assurance that EFW System design basis requirements will be met and does not affect EFW system configuration, setpoints, or reliability. These changes will not affect any accident initiation sequence and do not affect off site dose consequences of accidents that have been analyzed. The revised Accident Monitoring Instrumentation specification for the EFW flow instruments is needed to reflect the revised flowpath definition and does not change the intent or interpretation of this specification. The editorial changes included in this LCA [license change application] are intended to improve the clarity, consistency and readability of the TS[s], [and] do not change the intent or interpretation.

Therefore, operation of the facility in accordance with this proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. As a result of this change, no additional hardware is being added; and there will be no affect on EFW System design, operation as described in the SAR, or assumptions used in plant specific analyses. The requirement for three EFW Pumps and [associated] flowpaths to be operable for continuous plant operation is not affected by this change. Events involving the EFW System operation have been reviewed and determined to have no impact from these changes. The additional operability requirements, revised LCOs and surveillance standards, clarifications and changes to define EFW flowpath redundancy ensures minimum EFW component operability as credited in plant analyses. There are no changes included that could affect the plant beyond those accidents that have been evaluated. The editorial changes included in this LCA are intended to improve the clarity, consistency, and readability of the TS[s] and Bases, [and] do not change the intent or interpretation.

Therefore, operation of the facility in accordance with this proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

No. This change does not affect EFW System design or instrumentation setpoints. The requirement for three operable EFW pumps and associated flowpaths is not affected by this change. This change revises the Limiting Conditions of [for] Operation (LCOs) for the EFW System, revises the required actions, impose[s] additional required action times, and provide[s] clarification of the LCO and Surveillance Standards. The revised LCO requires that at least one flowpath to each OTSG must be operable. The 8 hour action time currently allowed for pump inoperability during surveillance testing is also applied to flowpath inoperability during testing. The revised LCO continues to require compensatory measures during EFW testing when HSPS is required to be operable and an OTSG is isolated, retaining the provision that EFW flowpath valves can be realigned promptly from their test mode to their operational alignment if EFW flow is needed. None of these changes affect a margin of safety. The revised Accident Monitoring Instrumentation specification for the EFW flow instruments is needed to reflect the revised flowpath definition and does not change the intent or interpretation of this specification. The editorial changes included in this LCA are intended to improve the clarity, consistency, and readability of the TS[s], [and] do not change the intent or interpretation.

Therefore, operation of the facility in accordance with this 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:* Edward J. Cullen, Jr., Esquire, Vice President, General Counsel and Secretary, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

*NRC Section Chief:* Joel T. Munday, Acting.

*Calvert Cliffs Nuclear Power Plant, Inc., Docket No. 50–318, Calvert Cliffs Nuclear Power Plant, Unit No. 2, Calvert County, Maryland*

*Date of amendment request:* November 19, 2001.

*Description of amendment request:* The amendment would revise the Technical Specification 5.5.16 to eliminate the requirement to perform post-modification containment integrated leakage rate testing following replacement of Unit 2 steam generators.

*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 steam generator replacement activities do not affect the containment structure or the actual containment liner. Access for the replacement steam generators as well as removal of the old steam generators will be through the equipment hatch. However, the outer shell of the steam generators, the inside containment portions of the main steam line, the feedwater lines, the auxiliary feedwater lines, and the steam generator blowdown lines are all part of the primary reactor containment boundary that will be impacted by the replacement activities.

Calvert Cliffs Nuclear Power Plant Technical Specification 5.5.16 states, ''A program shall be established to implement the leakage testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program,'' dated September 1995, including errata.'' Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program,'' endorses Nuclear Energy Institute (NEI)94–01, Revision 0 for methods acceptable to comply with the requirements of Option B. Prior to returning the Containment to operation, NEI 94–01 requires leakage rate testing (Type A testing or local leakage rate testing), following repairs and modification that affect the containment leakage integrity.

The affected area of the primary containment boundary is also part of the pressure boundary of an American Society of Mechanical Engineers (ASME) Class 2 component/piping system and, as such, the planned replacement of the steam generators are subject to the repair and replacement requirements of ASME Section XI. The ASME Section XI surface examination, volumetric examination, and system pressure test requirements are more stringent than the Appendix J, Option B testing requirements. The acceptance criteria for ASME Section XI system pressure testing of welded joints in ''zero leakage.'' In addition, the test pressure for the system pressure test will be approximately 17 times that of Appendix J, Option B test.

The objective of the Type A test is to assure the leak-tight integrity of the area affected by the modification. Although the leak test is in a direction reverse to that of the design basis accident environment, the ASME Section XI inspection and testing requirements more than fulfill the intent of the requirements of Appendix J, Option B with the exception of secondary side access manways. Section 9.2.1, NEI 94–01, Revision 0 allows reverse testing if justified. Section XI pressure test applies a sealing pressure to the secondary manway due to the inward door swing configuration. Hence, a Type B local leak rate test will be performed for the secondary manways. For all other affected components, reverse testing is justified since the acceptance criteria for ASME Section XI system pressure testing of welded joints is ''zero leakage,'' and the test pressure for the system pressure test will be approximately 17 times that of Type A test. Hence, the probability or consequences of design basis accidents previously evaluated are unchanged.

Therefore, the proposed revision to Technical Specification 5.5.16 to eliminate the requirement to perform post-modification containment integrated leakage rate testing following replacement of Unit 2 steam generators will 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.

The proposed revision does not involve a physical change to the plant and there are no changes to the operation of the plant that could introduce a new failure mode. As described above in Item 1, the objective of the Appendix J, Option B test is to assure the leak-tight integrity of the area affected by the modification. The ASME Section XI inspection and testing requirements are more stringent than the Appendix J, Option B testing requirements.

Therefore, the proposed revision to Technical Specification 5.5.16 to eliminate the requirement to perform post-modification containment leakage integrated rate testing following replacement of Unit 2 steam generators 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.

As described above in Item 1, the ASME Section XI surface examination, volumetric examination, and system pressure test requirements are more stringent than the Appendix J, Option B testing requirements. The acceptance criteria for ASME Section XI system pressure testing of welded joints is ''zero leakage.'' In addition, the test pressure for the system pressure test will be approximately 17 times that of Appendix J, Option B test.

Therefore, the proposed revision to Technical Specification 5.5.16 to eliminate the requirement to perform post-modification containment integrated leakage rate testing following replacement of Unit 2 steam generators 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:* Jay E. Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

*NRC Section Chief:* Joel Munday, Acting.

*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:* January 31, 2002.

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

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on September 28, 2001 (66 FR 49714). The licensee affirmed the applicability of the following NSHC determination in its application dated January 31, 2002.

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

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

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

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

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

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

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

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

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

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

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

*NRC Section Chief:* Joel Munday, Acting.

*Duke Energy Corporation, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina*

*Date of amendment request:* December 20, 2001.

*Description of amendment request:* The amendments would revise the Technical Specification (TS) 3.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation, and TS 3.3.5, Loss of Power Diesel Generator Start Instrumentation for Catawba Nuclear Station, Units 1 and 2. These amendments would modify the subject TS as summarized below.

1. Add a new MODE 3 operability requirement within ESFAS Function 5 (Turbine Trip and Feedwater Isolation) as shown on TS Table 3.3.2–1; reformat TS Table 3.3.2–1 in regard to ESFAS Function 5; modify identified Conditions and Required Actions applicable within ESFAS Function 5; and modify the content and footnotes applicable to ESFAS Functions 5 and 6 (Auxiliary Feedwater).

2. Delete ESFAS Functions 5e (Dog House Water Level—High High) and 5f (Turbine Trip and Feedwater Isolation, Trip of all Main Feedwater Pumps).

3. Modify the Conditions and Required Actions for ESFAS Function 6d (Auxiliary Feedwater, Loss of Offsite Power).

4. Modify the Conditions and Required Actions for ESFAS Function 6e (Auxiliary Feedwater, Trip of all Main Feedwater Pumps).

5. Modify the Conditions and Required Actions for ESFAS Function 6f (Auxiliary Feedwater, Auxiliary Feedwater Pump Train A and Train B Suction Transfer on Suction Pressure—Low).

6. Make an editorial change to ESFAS Function 8 (ESFAS Interlocks, $T_{avg}$—Low Low, P–12).

7. Add a new TS Surveillance Requirement (SR 3.3.2.12) for ESFAS Function 10 (Nuclear Service Water Suction Transfer—Low Pit Level).

8. Add a note to Condition A of TS 3.3.5 which allows one channel per bus to be bypassed for surveillance testing.

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

The following discussion is a summary of the evaluation of the changes contained in this proposed amendment against the 10 CFR 50.92(c) requirements to demonstrate that all three standards are satisfied. A no significant hazards consideration is indicated if operation of the facility in accordance with the proposed amendment would not:

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

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

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

First Standard

Implementation of this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. Approval of this amendment will have no effect on accident probabilities or consequences. For the proposed changes to Technical Specifications (TS) 3.3.2, Engineered Safety Features Actuation System (ESFAS); and 3.3.5, Loss of Power Diesel Generator Start Instrumentation; the equipment referenced in these TS is not accident initiating equipment. Therefore, there will be no impact on any accident probabilities caused by the NRC approval of this amendment. Additionally, since the design of the equipment is not being adversely modified by these proposed changes, there will be no impact on any accident consequences.

Second Standard

Implementation of this amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of the NRC approval of this license amendment request. No changes are being made to the plant which will introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators; therefore, no new accident types are being created.

Third Standard

Implementation of this amendment would not involve a significant reduction in a margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these fission product barriers will not be impacted by implementation of this proposed amendment. The equipment referenced in the proposed change to TS 3.3.2 and 3.3.5 will remain capable of performing as designed. No safety margins will be impacted.

Conclusion

Based upon the preceding discussion, Duke Energy Corporation has concluded that this proposed amendment does not involve a significant hazards consideration.

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

*Attorney for licensee:* Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

*NRC Section Chief:* Richard J. Laufer, Acting.

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

*Date of amendment request:* February 6, 2002.

*Description of amendment request:* The proposed amendment would remove the requirement for Main Steam Isolation Valve isolations on certain area temperatures from Technical Specifications Section 3.3.6.1, ''Primary Containment and Drywell Isolation Instrumentation.''

*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.

There is no credit taken in any licensing basis analysis for the main steam line isolation valve (MSIV) closure on the turbine area high temperature and there are no calculations that credit the subject isolation function as a mitigative feature. A review of Chapters 5, 6, 11, 12, and 15 of the USAR [Updated Safety Analysis Report] confirmed that the subject isolation function was not credited in any analysis for mitigating fuel cladding damage, mitigating challenges to vessel integrity, or mitigating dose to plant staff or the general public. This conclusion is consistent with the discussion of the function in the current Technical Specification [TS] Bases (B 3.3.6.1). Removing this requirement from the TS will allow the licensee to make changes to the design or function of the instrumentation provided the changes meet the 10 CFR 50.59 criteria. Entergy intends to make changes that will reduce unwarranted challenges to the MSIVs, associated isolation and actuation logic, and minimize the likelihood of an unwarranted plant transient due to increased ambient temperatures for reasons other than a steam leak. 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 removes the automatic MSIV isolation function associated with high temperatures in certain Turbine Building areas from the requirements of the Technical Specifications. Relocating requirements for this isolation function to licensee control does not introduce any new failure mechanisms or introduce any new accident precursors. Any subsequent changes to the design or function of the instrumentation must meet the criteria of 10 CFR 50.59.

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.

There is no credit taken in any licensing basis analysis for the main steam line isolation (MSIV closure) on the turbine area high temperature. Therefore, since the MSIV isolation function on the Turbine Building Area High Temperature is not credited as a mitigating feature in any analysis which establishes thermal limits, evaluates peak vessel pressure, evaluates peak containment/drywell pressure, or evaluates radiological consequences (on and off site), there is no adverse impact on any margin of safety.

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

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

The NRC staff has reviewed the licensee's analysis and, based on 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:* Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

*NRC Section Chief:* Robert A. Gramm.

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

*Date of amendment request:* January 31, 2002.

*Description of amendment request:* The proposed amendment would modify the Arkansas Nuclear One, Unit 2 (ANO–2) Facility Operating License (FOL) and Technical Specifications (TSs) to reorganize the Administrative Controls section (Section 6.0) to be consistent with NUREG–1432, ''Standard Technical Specifications Combustion Engineering Plants'' and provide consistency with the Arkansas Nuclear One, Unit 1 (ANO–1) TS Section 6.0. This change would result in moving several surveillance requirements, currently contained in the Surveillance Requirements section of the ANO–2 TSs, and programs contained in the FOL, to Section 6.0. The change would also result in the deletion of several TSs currently contained in Section 6.0. A Bases Control Program would also be added to Section 6.0. The TS actions related to the Control Room Ventilation System would also be modified as part of the proposed amendment. The ventilation system (emergency and air conditioning system) for the control room is shared with ANO–1 and, thus, the TSs for this system are maintained consistent between the units where appropriate.

*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 modifies the Administrative Controls section of the ANO–2 TSs to be consistent with NUREG–1432. [The requirements of ] 10 CFR 50.36, ''Technical Specifications'' defines the Administrative Controls section as follows: ''Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.'' Therefore, by definition the specifications contained in the Administrative Controls section are not specifications related to systems that are used to mitigate any types of accidents. The proposed changes to the Administrative Controls section therefore do not impact the ability of a plant system to perform its intended function.

The proposed changes to the Control Room Ventilation System specifications do not result in any type of plant modification to this system. The system's intended function is to provide heating, ventilation, and air conditioning to ensure a suitable environment for equipment and station operator comfort and safety.

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 will re-organize the ANO–2 Administrative Controls section and modify the actions related to the Control Room Ventilation System. The changes to the Administrative Controls section by definition of the type of specifications, which are included in the Administrative Controls section, will not create any new or different types of accidents.

The modifications to the Control Room Ventilation System specifications result in providing clarity to existing actions and the addition of new actions. The addition of the new actions results in consistency between the ANO–1 and ANO–2 TSs. No design changes are proposed to the Control Room Ventilation System.

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 changes result in the relocation of several surveillance requirements to the Administrative Controls section as well as the re-organization of the Administrative Controls Section of the ANO–2 TSs. In addition, clarification is added to the Control Room Ventilation System action statements that result in consistency between the ANO–1 and ANO–2 TSs. These changes do not affect the margin of safety.

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.

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

*Date of amendment request:* February 20, 2002.

*Description of amendment request:* The proposed amendment would revise Section 3.6.5 of the St. Lucie Unit 1 and 2 Technical Specifications (TS) to extend the allowed outage time for the Containment vacuum relief lines from 4 hours to 72 hours, in order to facilitate compliance with the Inservice Testing Program without placing the plants at risk for unnecessary shutdowns. The extended allowed outage time would provide sufficient time to perform the required surveillance tests and make any required adjustments on the

Containment vacuum relief valves. The proposed changes are consistent with NUREG–1432, ''Standard Technical Specifications for Combustion Engineering Plants.'' *Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not create any new system interactions and have no impact on operation or function of any system or equipment in a way that could cause an accident. The primary containment to annulus vacuum relief valves are part of the containment vacuum relief system and are not initiators of any events nor affect any accident initiators of any events previously analyzed in Chapters 6 or 15 of the UFSAR [Updated Final Safety Analysis Report].

The primary containment to annulus vacuum relief valves are designed to mitigate the consequences of an inadvertent containment spray system actuation during normal plant operation. The UFSAR analysis determined that with one of the two containment vacuum lines failed, the resultant peak calculated external pressure load on the containment was less than the design external pressure loading of 0.7 psi. These proposed changes do not affect any of the assumptions used in the analysis. Hence, the consequences of the design basis accident previously evaluated do not change.

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

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes do not alter the design, configuration, or method of operation of the plant. There is no change being made to the parameters within which the plant is operated. The setpoints at which the protective or mitigating actions are initiated are unaffected by this change. As such, no new failure modes are being introduced that would involve any potential initiating events that would create any new or different kind of accident.

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

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The proposed changes do not affect the bases used in or the results of the 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 impacted by the proposed change. The proposed change is acceptable because it assures at least one vacuum relief line will remain available in the event of a single failure. This further assures the ability to actuate upon demand for the purpose of mitigating the consequences of the design basis accident (inadvertent actuation of the containment spray system during normal operation). The remaining vacuum relief line provides sufficient vacuum relief capacity to prevent exceeding the design external pressure loading on containment of 0.7 psi.

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

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

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

*NRC Section Chief:* Richard P. Correia.

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

*Date of amendment requests:* February 22, 2002.

*Description of amendment requests:* The proposed amendments would relocate technical specifications (TSs) 3/4.9.6, ''Refueling Operations—Manipulator Crane Operability'' and TSs 3/4.9.7, ''Refueling Operations—Crane Travel—Spent Fuel Storage Pool Building,'' with associated Bases to the D. C. Cook updated final safety analysis report (UFSAR).

*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 of occurrence or consequences of an accident previously evaluated?

Response: No.

The proposed changes are administrative in nature in that they result in relocation of requirements from TS 3/4.9.6 and 3/4.9.7, with associated Bases, to the CNP UFSAR. Changes to the UFSAR are controlled by 10 CFR 50.59. Regulation 10 CFR 50.59 requires that NRC approval be obtained prior to any change to the UFSAR that would result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated. Accordingly, the relocation of requirements from TS 3/4.9.6 and 3/4.9.7, with associated Bases to the CNP UFSAR provides continued protection from changes involving unapproved increases in the probability of occurrence of an accident. The relocation of the requirements of TS 3/4.9.6 and 3/4.9.7 would not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, configuration of CNP or the manner in which it is operated. Therefore, the proposed change does not significantly increase the probability of occurrence of an accident previously evaluated.

The proposed change to relocate TS 3/4.9.6 and 3/4.9.7, with associated Bases to the CNP UFSAR does not impact the consequences of an accident because there is no effect on the structures, systems and components that mitigate the effects of an accident, or the manner in which they are operated. In accordance with 10 CFR 50.59, if any proposed change to the UFSAR results in more than a minimal increase in the consequences of an accident previously evaluated, NRC review and approval is required prior to the change being made. Accordingly, the relocation of requirements from TS 3/4.9.6 and 3/4.9.7, with associated Bases to the CNP UFSAR provides continued protection from changes involving unapproved increases in the probability of in the consequences of an accident. Therefore, the relocation of requirements will not affect offsite doses, and the consequences of an accident previously evaluated are not significantly increased.

The format changes improve the appearance of the affected pages but do not affect any requirements.

Therefore, the probability of occurrence and the consequences of an accident previously evaluated are not significantly increased.

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

Response: No.

The proposed change to relocate TS 3/4.9.6 and 3/4.9.7, with associated Bases, to the CNP UFSAR does not create new accident causal mechanisms. Plant operation will not be affected by the proposed change and no new failure modes will be created. Regulation 10 CFR 50.59 requires that NRC approval be obtained prior to any change to the UFSAR that would create the possibility of a new or different kind of accident from any accident previously evaluated. Accordingly, the relocation of requirements from TS 3/4.9.6 and 3/4.9.7, with associated Bases to the CNP UFSAR provides continued protection from unapproved changes involving new or different kinds of accidents.

The format changes improve the appearance of the affected pages but do not affect any requirements.

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

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

Response: No.

The proposed change to relocate the requirements from the TS to the UFSAR does not impact equipment design or operation and no changes are being made to the TS required safety limits, safety system settings, or any safety margins associated with TS 3/4.9.6 and 3/4.9.7. Changes to the UFSAR are controlled under the 10 CFR 50.59 process, which requires a safety evaluation to be performed. If any proposed change to the UFSAR results in a design basis limit for a fission product barrier, as described in the UFSAR, being exceeded or altered or results in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses, NRC review and approval will be required prior to the change being made. Accordingly, the relocation of requirements from TS 3/4.9.6 and 3/4.9.7, with associated Bases to the CNP UFSAR provides continued protection from changes involving a reduction in the margin of safety. The format changes improve the appearance of the affected pages but do not affect any requirements.

Therefore, there is no 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 requests involve no significant hazards consideration.

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

*NRC Section Chief:* William D. Reckley, Acting Section Chief.

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

*Date of amendment request:* August 16, 2001.

*Description of amendment request:* The proposed amendment would terminate license jurisdiction for a portion of the Maine Yankee Atomic Power Station (Maine Yankee) site, thereby releasing these lands from Facility Operating License No. DPR–36. In part, the release of these lands will facilitate the donation of a portion of this property to an environmental organization pursuant to a Federal Energy Regulatory Commission approved settlement between Maine Yankee Atomic Power Company and its ratepayers. The lands donated will be used to create a nature preserve and an environmental education center.

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

The proposed change does not:

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

The requested license amendment involves release of land presently considered part of the Maine Yankee plant site under license DPR–36. The land in question is not used for any licensed activities. No radiological materials have historically been used on this land and the land will not be used to support ongoing decommissioning operations and activities.

Most of the land to be released is outside the Exclusion Area Boundary and therefore is not affected by the consequences of any postulated accident. A small portion of the land is within the Exclusion Area Boundary. Maine Yankee will retain sufficient control over activities performed within this land through rights granted in the legal land conveyance documents to ensure that there is no impact on consequences from postulated accidents. Therefore, the release of the land from the [10 CFR] Part 50 license will not increase the probability or the consequences of an accident previously evaluated.

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

The requested amendment involves release of land presently considered part of the Maine Yankee plant site under license DPR–36. The land is not used for any licensed activities or decommissioning operations. The proposed action does not affect plant systems, structures or components in any way. The requested release of the land does not 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 margin of safety defined in the statements of consideration for the final rule on the Radiological Criteria for License Termination is described as the margin between the 100 mrem/yr public dose limit established in 10 CFR 20.1301 for licensed operation and the 25 mrem/yr dose limit to the average member of the critical group at a site considered acceptable for unrestricted use. This margin of safety accounts for the potential effect of multiple sources of radiation exposure to the critical group. Additionally, the State of Maine, through legislation, has imposed a 10 mrem/yr all pathways limit, with no more than 4 mrem/yr attributable to drinking water sources. Since the area is non-impacted, there will be no additional dose to the average member of the critical group. Furthermore, the survey results described in Attachment III [of the August 16, 2001, application] demonstrate that residual radioactivity, if any, in the area is indistinguishable from background. Therefore, this proposed license 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 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:* Robert A. Gramm.

*North Atlantic Energy Service Corporation, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire*

*Date of amendment request:* February 21, 2002.

*Description of amendment request:* The proposed amendment would relocate specific pressure, differential pressure, and flow values, as well as specific test methods, associated with certain Engineered Safeguards Features (ESF) pumps from the Technical Specifications to the Seabrook Station Technical Requirements Manual.

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

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

The proposed changes to relocate the specific ESF pump pressure and flow criteria in the aforementioned Technical Specifications surveillance requirements to the Seabrook Station Technical Requirements Manual are administrative in nature and do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, configuration of the facility or the manner in which it is operated. The proposed changes do not alter or prevent the ability or structures, systems, or components to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Seabrook Station Updated Final Safety Analysis Report (UFSAR).

The subject surveillance requirement criteria relocated to the Seabrook Station Technical Requirements Manual will continue to be administratively controlled. The Seabrook Station Technical Requirements is a licensee-controlled document, which contains certain technical requirements and is the implementing manual for the Technical Specification Improvement Program. Changes to these requirements are reviewed and approved in accordance with Seabrook Station Technical Specifications, Section 6.7.1.i, and as outlined in the Seabrook Station Technical Requirements Manual.

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

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

The proposed changes do not alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated. There are no changes to the source term or radiological release assumptions used in evaluating the radiological consequences in the Seabrook Station UFSAR. The proposed changes have no adverse impact on component or system interactions. The proposed changes will not adversely degrade the ability of systems, structures and components important to safety to perform their safety function nor change the response of any system, structure or component important to safety as described in the UFSAR. The proposed changes are administrative in nature and do not change the level of programmatic and procedural details of assuring operation of the facility in a safe manner. Since there are no changes to the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated and surveilled, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

There is no adverse impact on equipment design or operation and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. The proposed changes are administrative in nature and do not reduce the level of programmatic or procedural controls associated with the activities presently performed via the aforementioned surveillance requirements.

Future changes to the subject technical requirements will be reviewed and approved in accordance with Seabrook Station Technical Specifications, Section 6.7, and as outlined in North Atlantic [Energy Service Corporation]'s programs. Specifically, changes to the Seabrook Station Technical Requirements Manual require an evaluation pursuant to the provisions of 10 CFR 50.59 and review and approval by the Station Operation Review Committee (SORC) prior to implementation.

Therefore, relocation of the specific pump pressure and flow criteria contained in the aforementioned Technical Specifications Surveillance Requirements to the Seabrook Station Technical Requirements Manual does not involve a significant reduction in the margin of safety provided in the existing specifications.

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

*Attorney for licensee:* William J. Quinlan, Esq., Assistant General Counsel, Northeast Utilities Service Company, PO Box 270, Hartford CT 06141–0270.

*NRC Section Chief:* James W. Clifford.

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

*Date of amendment request:* January 11, 2002.

*Description of amendment request:* The proposed amendments would revise Technical Specification (TS) 3.6.4, "Containment Pressure," to reduce the maximum allowable pressure from 3 pounds per square inch gauge (psig) to 2 psig. The licensee requests these proposed amendments to address a non-conservatism that was identified during reviews of the Point Beach, Units 1 and 2, accident analyses.

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

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

The operability of containment ensures that radionuclides are contained within allowable limits during and following all credible accident conditions. The inoperability or failure of containment is not a design basis accident initiator or precursor. Therefore, the probability of an accident previously evaluated will not be significantly increased as a result of the proposed change. Because design limitations continue to be met and the integrity of the containment system pressure boundary is not challenged, the assumptions employed in the calculation of the offsite radiological doses remain valid. In addition, the radiological consequence analysis for the main steam line break (MSLB) is performed assuming the MSLB is outside of the containment. Therefore, the operability of the containment structure does not affect the results of the offsite dose or control room dose consequences.

Therefore, the consequences of an accident previously evaluated will not be significantly increased as a result of the proposed change.

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

The possibility for a new or different type of accident from any accident previously evaluated is not created as a result of this amendment. The evaluation of the effects of the proposed changes indicate that all design standards and applicable safety criteria limits are met. These changes, therefore, do not cause the initiation of any new or different accident nor create any new failure mechanisms.

Equipment important to safety will continue to operate as designed. Component integrity is not challenged. The changes do not result in any event previously deemed incredible being made credible. The changes do not result in more adverse conditions or result in any increase in the challenges to safety systems. Therefore, operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not create the possibility of a new or different type of accident from any accident previously evaluated.

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

The containment functions to mitigate the effects of accidents. There are no new or significant changes to the initial conditions contributing to accident severity or consequences. The proposed modification will not otherwise affect the plant protective boundaries, will not cause a release of fission products to the public, nor will it degrade the performance of any other SSCs [structures, systems, and components] important to safety. Reducing the maximum allowed containment pressure limit is conservative in that it reduces the peak containment pressure that could result in the event of an accident. Therefore, reducing the maximum allowed containment pressure limit will not reduce the margin of safety. The added conservatism provides improvement to the design pressure margin resulting from the proposed change and will enhance protection against conditions resulting from a design basis accident, which will therefore provide a net benefit to radiological health and reactor safety.

Conclusion

Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not result in a significant increase in the probability or consequences of any accident previously analyzed; will not result in a new or different kind of accident from any accident previously analyzed; and, does not result in a significant reduction in any margin of safety. Therefore, operation of PBNP [Point Beach Nuclear Plant] in accordance with the proposed amendments does not result in a significant hazards determination.

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

*Attorney for licensee:* John H. O'Neill, Jr., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

*NRC Section Chief:* William Reckley, Acting.

*Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California*

*Date of amendment requests:* February 13, 2002.

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

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on September 28, 2001 (66 FR 49714). The licensee affirmed the applicability of the following NSHC determination in its application dated February 13, 2002.

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

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

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

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

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

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

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

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

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

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

*Attorney for licensee:* Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

*NRC Section Chief:* Stephen Dembek.

*Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California*

*Date of amendment requests:* February 22, 2002.

*Description of amendment requests:* The proposed amendment would revise technical specifications (TSs) for San Onofre Nuclear Generating Station (SONGS), Units 2 and 3 relating to spent fuel storage. Specifically, TS 3.7.17, ''Fuel Storage Pool Boron Concentration'', TS 3.7.18, ''Spent Fuel Assembly Storage'', and TS 4.3, ''Fuel Storage'' would be revised to remove credit for use of Boraflex, and to take credit for soluble boron, and to increase the required concentration of soluble boron in the spent fuel storage pool. Additionally, new TS 5.5.2.16, ''Fuel Storage Program'' would be added to create a TS to control the Fuel Storage Program.

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

No.

Dropped Fuel Assembly

There is no significant increase in the probability of a fuel assembly drop accident in the spent fuel pool when assuming a complete loss of the Boraflex panels in the spent fuel pool racks and considering the presence of soluble boron in the spent fuel pool water for criticality control.

The presence of soluble boron in the spent fuel pool water for criticality control does not increase the probability of a fuel assembly drop accident. The handling of the fuel assemblies in the spent fuel pool has always been performed in borated water, and the quantity of Boraflex remaining in the racks has no affect on the probability of such a drop accident.

Southern California Edison (SCE) has performed a criticality analysis which shows that the consequences of a fuel assembly drop accident in the spent fuel pool are not affected when considering a complete loss of the Boraflex in the spent fuel racks and the presence of soluble boron. The rack $K_{eff}$ remains less than or equal to 0.95.

Fuel Misloading

There is no significant increase in the probability of the accidental misloading of spent fuel assemblies into the spent fuel racks when assuming a complete loss of the Boraflex panels and considering the presence of soluble boron in the pool water for criticality control. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures and will be in accordance with the Technical Specification Section 5.5.2.16, ''Fuel Storage Program,'' which will specify spent fuel rack storage configuration limitations.

There is no increase in the consequences of the accidental misloading of a spent fuel assembly into the spent fuel racks. The criticality analysis, performed by SCE, demonstrates that the pool $K_{eff}$ will be maintained less than or equal to 0.95 following an accidental misloading by the boron concentration of the pool. The proposed Technical Specification 3.7.17 will ensure that an adequate spent fuel pool boron concentration is maintained.

Significant Change in Spent Fuel Pool Temperature

There is no significant increase in the probability of either the loss of normal cooling to the spent fuel pool water or a decrease in pool water temperature from a large emergency makeup when assuming a complete loss of the Boraflex panels and considering the presence of soluble boron in the spent fuel pool water. A high concentration [> 2000 parts per million (ppm)] of soluble boron has always been maintained in the spent fuel pool water. The proposed minimum boron concentration of 2000 ppm in Technical Specification 3.7.17 will ensure that an adequate spent fuel pool concentration is maintained in the spent fuel pools.

A loss of normal cooling to the spent fuel pool water causes an increase in the temperature of the water passing through the stored fuel assemblies. This causes a decrease in water density, and when coupled with the assumption of a complete loss of Boraflex, may result in a positive reactivity addition. However, the additional negative reactivity provided by the boron concentration limit in the proposed Technical Specification 3.7.17 will compensate for the increased reactivity which could result from a loss of spent fuel pool cooling. Because adequate soluble boron will be maintained in the spent fuel pool water to maintain $K_{eff}$ less than or equal to 0.95, the consequences of a loss of normal cooling to the spent fuel pool will not be increased.

A decrease in pool water temperature causes an increase in water density and may result in an increase in reactivity when the Boraflex panels are present in the racks. However, the additional negative reactivity provided by the boron concentration limit in the proposed Technical Specification 3.7.17, determined based on the conservative assumption of a complete loss of the Boraflex, will compensate for the increased reactivity which could result from a decrease in pool water temperature.

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

No.

Criticality accidents in the spent fuel pool are not new or different. They have been analyzed in the Updated Final Safety Analysis Report (UFSAR) and in previous submittals to the Nuclear Regulatory Commission (NRC). Specific accidents considered and evaluated include fuel assembly drop, fuel assembly misloading in the racks, and spent fuel pool water temperature changes.

The possibility for creating a new or different kind of accident is not credible. Neither Boraflex or soluble boron are accident initiators. The proposed change takes credit for soluble boron in the spent fuel pool while maintaining the necessary margin of safety. Because soluble boron has always been present in the spent fuel pool, a dilution of the spent fuel pool soluble boron has always been a possibility. However, this accident was not considered credible. For this proposed amendment, SCE performed a spent fuel pool dilution analysis, which demonstrated that a dilution of the boron concentration in the spent fuel pool water which could increase the rack $K_{eff}$ to greater than 0.95 (constituting a reduction of the required margin to criticality) is not a credible event. The requirement to maintain boron concentration in the spent fuel pool water for reactivity control will have no effect on normal pool operations and maintenance. There are no changes in equipment design or in plant configuration. This new requirement will not result in the installation of any new equipment or modification of any existing equipment.

Therefore, the proposed change will not result in 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 Technical Specification changes proposed by this License Amendment request and the resulting spent fuel storage operation limits will provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. Those limits are based on a San Onofre Nuclear Generating Station (SONGS), Units 2 and 3 plant specific analysis performed in accordance with a methodology previously approved by the NRC.

The proposed change takes partial credit for soluble boron in the spent fuel pool. SCE's analyses show that spent fuel storage requirements meet the following NRC acceptance criteria for preventing criticality outside the reactor:

(1) The neutron multiplication factor, $K_{eff}$, including all uncertainties, shall be less than 1.0 when flooded with unborated water, and,

(2) The neutron multiplication factor, $K_{eff}$, including all uncertainties, shall be less than or equal to 0.95 when flooded with borated water.

The criticality analysis utilized credit for soluble boron to ensure $K_{eff}$ will be less than or equal to 0.95 under normal circumstances, and storage configurations have been defined using a 95/95 $K_{eff}$ calculation to ensure that the spent fuel rack will be less than 1.0 with no soluble boron. Soluble boron credit is used to provide safety margin by maintaining $K_{eff}$ less than or equal to 0.95 including uncertainties, tolerances and accident conditions in the presence of spent fuel pool soluble boron. The loss of a substantial amount of soluble boron from the spent fuel pool water which could lead to $K_{eff}$ exceeding 0.95 has been evaluated and shown to not be credible.

Also, the spent fuel rack $K_{eff}$ will remain less than 1.0 with the spent fuel pool flooded with unborated water.

Decay heat, radiological effects, and seismic loads are unchanged by the absence of Boraflex.

Therefore, the proposed change does not involve a significant reduction in the plant's 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:* Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

*NRC Section Chief:* Stephen Dembek.

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

*Date of amendment request:* January 28, 2002.

*Description of amendment request:* The proposed amendment revises Technical Specifications (TS) 4.4.5.3a, ''Steam Generator Surveillance Requirements,'' inservice inspection frequency requirements for Unit 1 immediately after the first refueling outage (1RE09) and Unit 2 after the second refueling outage (2RE10). The change would allow a 40-month inspection interval after one inspection resulting in C–1 classification, rather than two consecutive inspection resulting in C–1 classification. The change is proposed to eliminate steam generator inspections, which will result in significant dose, schedule and cost savings.

*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.

There is no direct increase in SG [steam generator] leakage because the proposed change does not alter the plant design. The scope of inspections performed during 1RE10, the first refueling outage following SG replacement, exceeded the TS requirements for the first two refueling outages after replacement combined. That is, more tubes were inspected than were required by TS. Currently, South Texas Project Unit 1 does not have an active SG damage mechanism and will meet the current industry examination guidelines without performing inspections during the next refueling outage. The results of the Condition Monitoring Assessment after 1RE10 demonstrated that all performance criteria were met during 1RE10. The results of the 1RE10 Operational Assessment show that all performance criteria will be met over the proposed operating period. The results from 2RE10 inspections are expected to be the same. 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 will not alter any plant design basis or postulated accident resulting from potential SG tube degradation. The scope of inspections performed during 1RE10 and planned for 2RE10, the first refueling outage for each unit following SG replacement, significantly exceed the TS requirements for the scope of the first two refueling outages after SG replacement combined.

The proposed change does not affect the design of the SGs, the method of operation, or reactor coolant chemistry controls. No new equipment is being introduced and installed equipment is not being operated in a new or different manner. The proposed change involves a one-time extension to the SG tube inservice inspection frequency, and therefore will not give rise to new failure modes. In addition, the proposed change does not impact any other plant system or components. 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.

Steam generator tube integrity is a function of design, environment, and current physical condition. Extending the SG tube inservice inspection frequency by one operating cycle will not alter the function or design of the SGs. Inspections conducted prior to placing the SGs into service (pre-service inspections) and inspection during the first refueling outage following SG replacement demonstrate that the SGs do not have fabrication damage or an active damage mechanism. The scope of those inspections significantly exceeded those required by the TS. These inspection results were comparable to similar inspection results for the same model of RSGs [replacement steam generators] installed at other plants, and subsequent inspections at those plants yielded results that support this extension request. The improved design of the replacement SGs also provides reasonable assurance that significant tube degradation is not likely to occur over the proposed operating period. 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 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:* Morgan Lewis, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

*NRC Section Chief:* Robert A. Gramm.

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

*Date of amendment request:* February 18, 2002.

*Description of amendment request:* The proposed amendment revises Technical Specification (TS) 3/4.6.1.7, ''Containment Ventilation System,'' to extend the intervals between operability tests of the normal and supplementary containment purge valves.

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

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

Response: No.

Operability and leakage control effectiveness of the containment purge isolation valves have no effect on whether or not an accident occurs. Consequently, increasing the interval between surveillances of isolation valve effectiveness does not involve a significant increase in the probability of an accident previously evaluated.

The consequences of a non-isolated reactor containment building at the time of a fuel-handling accident or LOCA [loss-of-coolant accident] is release of radionuclides to the environment. Analyses have conservatively assumed that a purge system line is open at the time of an accident, and release to the environment continues until the isolation valves are closed. In addition, LOCA analyses assume containment leakage of 0.3 percent per day for the first 24-hours and 0.15 percent per day thereafter. Consequently, increasing the interval between surveillances of isolation valve effectiveness 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 previously evaluated?

Response: No.

The proposed changes do not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. The function of the containment purge systems is not altered by this change. Therefore, this proposed change does not create the possibility of an accident of a different kind than previously evaluated.

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

Response: No.

This proposed change only increases the interval between surveillance tests of the containment purge valves. Analyses have conservatively assumed that the normal purge valves are open at the time of a fuel handling accident, and that purging by the supplementary purge system is in progress at the time of a loss of coolant accident. In addition, LOCA analyses assume containment leakage of 0.3 percent per day for the first 24-hours and 0.15 percent per day thereafter. The radiological consequences of both a fuel handling accident and a LOCA are unchanged and remain within the 10 CFR 100 limits. 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 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:* Morgan Lewis, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

*NRC Section Chief:* Robert A. Gramm.

*Vermont Yankee Nuclear Power Corporation, Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont*

*Date of amendment request:* June 21, 2001, as supplemented on February 8, 2002.

*Description of amendment request:* This amendment request proposes to revise the control rod block instrumentation requirements contained in Technical Specification (TS) 2.1.B, Figure 2.1.1, and Tables 3.2.5 and 4.2.5. Some of the control rod block trip functions are being relocated to the Vermont Yankee Technical Requirements Manual and some of the requirements for the retained trip functions are being clarified. Two trip functions are added to the TSs and Note 9 to Table 3.2.5 is changed to reflect one or two Rod Block Monitor channels inoperable. This proposed no significant hazards consideration determination replaces in its entirety the notice published in the **Federal Register** on July 25, 2001 (66 FR 38769).

*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 operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The relocated trip functions are not assumed as initial conditions for, nor are they credited in the mitigation of, any design basis accident or transient previously evaluated. Since reactor operation with these revised and relocated Specifications is fundamentally unchanged, no design or analytical acceptance criteria will be exceeded. As such, this change does not impact initiators of analyzed events, or the analyzed mitigation of design basis accident or transient events.

More stringent requirements that ensure operability of equipment and purely administrative changes do not affect the initiation of any event, nor do they negatively impact the mitigation of any event. The addition of remedial actions to address a condition when both channels of the Rod Block Monitor (RBM) are inoperable also ensures that the RBM function is met. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

None of the proposed changes affects any parameters or conditions that could contribute to the initiation of any accident. No new accident modes are created since plant operation is unchanged in that required protective features remain operable. No safety-related equipment or safety functions are altered as a result of these changes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

This change does not impact plant equipment, nor does it involve operation with loss of any safety function. There are no changes being made to safety limits or safety system settings that would adversely affect plant safety as a result of the proposed changes. Since the changes have no effect on any safety analysis assumptions or initial conditions, the margins of safety in the safety analyses are maintained. In addition, administrative changes that do not change technical requirements or meaning, and the imposition of more stringent or equivalent remedial requirements to ensure operability, have no negative impact on margins of safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

*Attorney for licensee:* Mr. David R. Lewis, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

*NRC Section Chief:* James W. Clifford.

## Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

*Date of application for amendments:* September 11, 2001.

*Brief description of amendments:* The amendments revise Technical Specification Section 3.9.5, ''Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level,'' by adding a note that allows one SDC loop to be inoperable for a period of 2 hours provided the other loop is operable while in Mode 6.

*Date of issuance:* March 1, 2002.

*Effective date:* March 1, 2002, and shall be implemented within 45 days of the date of issuance.

*Amendment Nos.:* Unit 1–139, Unit 2–139, Unit 3–139.

*Facility Operating License Nos. NPF–41, NPF–51, and NPF–74:* The amendments revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** November 28, 2001 (66 FR 59501).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 2002.

No significant hazards consideration comments received: No.

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

*Date of application for amendments:* November 19, 2001.

*Brief description of amendments:* The amendments authorized revisions to the Calvert Cliffs Nuclear Power Plant Updated Final Safety Analysis Report to incorporate revisions to the loss of feedwater flow analysis.

*Date of issuance:* February 26, 2002.

*Effective date:* As of the date of issuance and shall be implemented in conformance with the scheduling requirements specified in 10 CFR 50.71e.

*Amendment Nos.:* 248, 224.

*Renewed Facility Operating License Nos. DPR–53 and DPR–69:* Amendments revised Appendix C of the licenses.

*Date of initial notice in* ***Federal Register:*** January 8, 2002 (67 FR 925).

The Commission's related evaluation of these amendments is contained in a

Safety Evaluation dated February 26, 2002.

No significant hazards consideration comments received: No.

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

*Date of application for amendments:* December 20, 2000, as supplemented on July 12, 2001.

*Brief description of amendments:* The amendments revise the Technical Specifications to incorporate changes required to support operation with replacement steam generators.

*Date of issuance:* March 1, 2002.

*Effective date:* As of the date of issuance and shall be implemented prior to restart following replacement of the steam generators.

*Amendment Nos.:* 249 and 225.

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

*Date of initial notice in **Federal Register:*** March 7, 2001 (66 FR 13799).

The July 12, 2001, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated March 1, 2002.

No significant hazards consideration comments received: No.

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

*Date of application for amendments:* July 27, 2001.

*Brief description of amendments:* The amendments modify the conditions and required actions for the control room emergency ventilation system (CREVS) and control room emergency temperature system (CRETS) of Technical Specifications (TS) 3.7.8 and 3.7.9. A Note is added to TS 3.7.8 and the Note for TS 3.7.9 is revised to specify train operability requirements during the movement of irradiated fuel assemblies.

*Date of issuance:* March 4, 2002.

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

*Amendment Nos.:* 250, 226.

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

*Date of initial notice in **Federal Register:*** September 5, 2001 (66 FR 46475).

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated March 4, 2002.

*No significant hazards consideration comments received:* No.

*Duke Energy Corporation, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, (CNS) Units 1 and 2, York County, South Carolina*

*Date of application for amendments:* August 6, 2001.

*Brief description of amendments:* The amendments revised the Technical Specifications (TS) by decreasing the CNS Unit 1 Overtemperature Delta Temperature Allowable Value and the CNS Units 1 and 2 Overpower Delta Temperature Allowable Values in TS Table 3.3.1–1. In addition, the amendments make two minor editorial changes in the TS Table of Contents and Bases Page 3.3.1–10.

*Date of issuance:* February 26, 2002.

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

*Amendment Nos.:* 195/188.

*Facility Operating License Nos. NPF–35 and NPF–52:* Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 22, 2002 (67 FR 2920). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 26, 2002.

No significant hazards consideration comments received: No.

*Duke Energy Corporation, Docket No. 50–287, Oconee Nuclear Station, Unit 3, Oconee County, South Carolina*

*Date of application of amendments:* March 5, 2001, as supplemented by letter dated September 4, 2001.

*Brief description of amendments:* The amendment revised the Technical Specifications to allow a one-time extension to the interval for conducting the 10 CFR part 50, Appendix J containment integrated leak rate test.

*Date of Issuance:* February 28, 2002.

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

*Amendment No.:* 321.

*Renewed Facility Operating License No. DPR–55:* Amendment revised the Technical Specification.

*Date of initial notice in **Federal Register:*** October 3, 2001 (66 FR 50466). The supplement dated September 4, 2001, provided clarifying information that did not change the scope of the March 5, 2001, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 28, 2002.

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* January 31, 2002.

*Brief description of amendment:* The amendment revised the Technical Specifications by replacing the peak linear heat rate safety limit with a peak fuel centerline temperature safety limit.

*Date of issuance:* March 4, 2002.

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

*Amendment No.:* 238.

*Facility Operating License No. NPF–6:* Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 11, 2002 (67 FR 6279). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 4, 2002.

No significant hazards consideration comments received: No.

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

*Date of amendment request:* July 10, 2001, as supplemented by letter dated December 20, 2001.

*Brief description of amendment:* Technical Specification (TS) Surveillance Requirement (SR) 4.8.1.1.2.e requires certain emergency diesel generator (EDG) surveillances be performed during shutdown. This change modifies this SR to allow performance of specific surveillances during any mode of plant operation. This provides the flexibility in the scheduling of testing activities consistent with online maintenance activities and improves EDG availability during plant shutdown periods.

*Date of issuance:* February 26, 2002.

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

*Amendment No.:* 180.

*Facility Operating License No. NPF–38:* The amendment revised the TS.

*Date of initial notice in **Federal Register:*** August 22, 2001 (66 FR 44168).

The December 20, 2001, supplemental letter contained clarifying information that did not change the scope of the July 10, 2001, application nor the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

*Date of amendment request:* January 31, 2002.

*Brief description of amendment:* The amendment replaces the Technical Specification (TS) Safety Limit 2.1.1.2, ''Peak Linear Heat Rate,'' (PLHR) with a Peak Fuel Centerline Temperature Safety Limit and updates the Index accordingly. The associated TS Bases changes have been made to appropriately reflect the proposed new Safety Limit.

*Date of issuance:* March 5, 2002.

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

*Amendment No.:* 181.

*Facility Operating License No. NPF–38:* The amendment revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** February 11, 2002 (67 FR 6281).

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

No significant hazards consideration comments received: No.

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

*Date of application for amendments:* May 14, 2001.

*Brief description of amendments:* The amendments deleted Technical Specification (TS) Figures 5.1–1, ''Site Area Map''; and 5.1–2, ''Plant Area Map''; and replaced TS 5.1, ''Site,'' with a site location description. Conforming changes also deleted TS 5.1.1, ''Exclusion Area''; TS 5.1.2, ''Low Population Zone''; and TS 5.1.3, ''Map Defining Unrestricted Areas and Site Boundary for Radioactive Gaseous and Liquid Effluents''; from TS 5.1 and the TS Index.

*Date of issuance:* February 12, 2002.

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

*Amendment Nos:* 219 and 213.

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

*Date of initial notice in* ***Federal Register:*** June 27, 2001 (66 FR 34284). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 12, 2002.

No significant hazards consideration comments received: No.

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

*Date of application for amendments:* July 18, 2001.

*Brief description of amendments:* The amendments revised Turkey Point Units 3 and 4 Technical Specifications, Section 6.0, ''Administrative Controls.'' The revision consists of changing the title of the corporate executive responsible for overall nuclear plant safety from ''President—Nuclear Division'' to ''Chief Nuclear Officer.''

*Date of issuance:* February 21, 2002.

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

*Amendment Nos:* 220 and 214.

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

*Date of initial notice in* ***Federal Register:*** August 8, 2001 (66 FR 41622). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 21, 2002.

No significant hazards consideration comments received: No.

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

*Date of application for amendments:* November 1, 2001.

*Brief description of amendments:* The amendments revise technical specification (TS) surveillance requirements (SR) 4.8.2.3.2.c.2 and 4.8.2.5.2.c.2 and associated TS bases concerning the safety-related batteries to make them more consistent with the Westinghouse Standard TSs.

*Date of issuance:* February 26, 2002.

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

*Amendment Nos.:* 266 and 247.

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

*Date of initial notice in* ***Federal Register:*** December 12, 2001 (66 FR 64296). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 26, 2002.

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* October 5, 2001, as revised on January 4, 2002.

*Brief description of amendment:* The amendment imposes a new license condition in Operating License NPF–69 to approve a change in the licensing basis regarding post-safety-injection hydrogen monitoring. Specifically, the amendment changes the permissible delay from 30 minutes to 90 minutes.

*Date of issuance:* February 25, 2002.

*Effective date:* As of the date of issuance to be implemented during Refueling Outage 8.

*Amendment No.:* 102.

*Facility Operating License No. NPF:–69* Amendment revises the the operating license.

*Date of initial notice in* ***Federal Register:*** January 22, 2002 (67 FR 2925). The staff's related evaluation of the amendment is contained in a Safety Evaluation dated February 25, 2002.

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* March 29, 2001, as supplemented October 30, 2001.

*Brief description of amendment:* The amendment revised the Technical Specifications (TS), Section 3.8.5, ''DC [Direct Current] Sources—Shutdown,'' restoring the operability requirement to what it was before the TS was converted to the Improved Standard Technical Specifications format (i.e., Amendment No. 91).

*Date of issuance:* March 1, 2002.

*Effective date:* As of the date of issuance, to be implemented prior to Refueling Outage 8.

*Amendment No.:* 103.

*Facility Operating License No. NPF–69:* Amendment revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** May 30, 2001 (66 FR 29359).

The licensee's October 30, 2001, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The staff's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 2002.

*No significant hazards consideration comments received:* No.

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

*Date of application for amendment*: August 15, 2001.

*Brief description of amendment*: The amendment revised the Technical Specifications (TS) to extend the channel calibration surveillance frequency for the automatic depressurization system timers from 18 months to 24 months.

*Date of issuance:* February 26, 2002.

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

*Amendment No.:* 245.

*Facility Operating License No. DPR–49*: The amendment revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** October 3, 2001 (66 FR 50469). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 26, 2002.

No significant hazards consideration comments received: No.

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

*Date of application for amendment*: March 2, 2001, as supplemented March 29, September 14, and December 27, 2001.

*Brief description of amendment*: The amendment changes the Technical Specifications to increase the limits on stored fuel enrichments and provide other more flexible fuel loading constraints for the storage racks for new and spent fuel.

*Date of issuance:* February 26, 2002.

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

*Amendment No.:* 207.

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

*Date of initial notice in* ***Federal Register:*** June 1, 2001 (66 FR 29844). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 26, 2002.

No significant hazards consideration comments received: No.

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

*Date of amendment request:* November 21, 2001.

*Brief description of amendment:* The amendment adds three topical report references to Technical Specification (TS) 5.9.5, ''Core Operating Limits Report.''

*Date of issuance:* March 4, 2002.

*Effective date:* March 4, 2002, to be implemented within 60 days from the date of issuance.

*Amendment No.:* 203.

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

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 4, 2002.

No significant hazards consideration comments received: No.

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

*Date of application for amendments:* October 18, 2001, as supplemented February 5, 2002.

*Brief description of amendments:* The amendments revised the Technical Specifications surveillance requirement 3.4.3.1 for testing of the main steam safety relief valves to permit the setpoint tolerance for ''as-found'' testing to be changed from ±1 percent to ±3 percent. An editorial change will also be made to remove a note regarding an associated relief request.

*Date of issuance:* March 7, 2002.

*Effective date:* As of the date of issuance, and shall be implemented during the spring 2002, and spring 2003, refueling and inspection outages for Units 1 and 2, respectively.

*Amendment Nos.:* 201, 175.

*Facility Operating License Nos. NPF–14 and NPF–22:* The amendments revised the TSs.

*Date of initial notice in* ***Federal Register:*** November 28, 2001 (66 FR 59511).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 7, 2002.

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* March 5, 2001, as supplemented on December 17, 2001.

*Brief description of amendment:* The amendment revises License Condition 2.E in the Facility Operating License (FOL) to reflect Nuclear Regulatory Commission (NRC) staff approval of a change to the Salem-Hope Creek Security Plan and the Salem-Hope Creek Security Training and Qualification Plan. The specific change reviewed and approved by the NRC staff will allow illumination levels to be maintained at a minimum of 0.2 footcandle in the isolation zone while allowing lighting in the remainder of the protected area to be sufficient as determined by the licensee, rather than requiring a minimum 0.2 footcandle illumination level in the entire protected area.

*Date of issuance:* February 22, 2002.

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

*Amendment No.:* 138.

*Facility Operating License No. NPF–57:* This amendment revised the FOL.

*Date of initial notice in* ***Federal Register:*** July 11, 2001 (66 FR 36343). The letter dated December 17, 2001, withdrew a portion of the March 5, 2001, application which would have changed the escort requirements for vehicles in the protected area. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 22, 2002.

No significant hazards consideration comments received: No.

*PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey*

*Date of application for amendments:* March 5, 2001, as supplemented on December 17, 2001.

*Brief description of amendments:* The amendments revise License Condition 2.E in each of the respective Facility Operating Licenses (FOLs) to reflect Nuclear Regulatory Commission (NRC) staff approval of a change to the Salem-Hope Creek Security Plan and the Salem-Hope Creek Security Training and Qualification Plan. The specific change reviewed and approved by the NRC staff will allow illumination levels to be maintained at a minimum of 0.2 footcandle in the isolation zone while allowing lighting in the remainder of the protected area to be sufficient as determined by the licensee, rather than requiring a minimum 0.2 footcandle illumination level in the entire protected area.

*Date of issuance:* February 22, 2002.

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

*Amendment Nos.:* 250 and 230.

*Facility Operating License Nos. DPR–70 and DPR–75:* The amendments revised each of the respective FOLs.

*Date of initial notice in* ***Federal Register:*** June 27, 2001 (66 FR 34288). The letter dated December 17, 2001, withdrew a portion of the March 5, 2001, application which would have changed the escort requirements for vehicles in the protected area. The

Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 22, 2002.

No significant hazards consideration comments received: No.

*South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina*

*Date of application for amendment:* June 19, 2001.

*Brief description of amendment:* This amendment revises V. C. Summer Technical Specification 3.4.6.2.f by increasing the allowable operational leakage rate for 23 of the 35 reactor coolant system pressure isolation valves listed in TS Table 3.4–1. This change implements a size-dependent allowable leakage rate of 0.5 gallon per minute per nominal inch of valve diameter, up to a maximum of 5 gallons per minute per valve.

*Date of issuance:* February 14, 2002.

*Effective date:* February 14, 2002.

*Amendment No.:* 154.

*Facility Operating License No. NPF–12:* Amendment revises the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** August 8, 2001 (66 FR 41626). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 14, 2002.

No significant hazards consideration comments received: No.

*South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina*

*Date of application for amendment:* June 19, 2001.

*Brief description of amendment:* This amendment revises Technical Specifications (TS) Table 3.7–1 by lowering the maximum allowable power range neutron flux high setpoints when one or more main steam line safety valves are inoperable. The Bases for TS 3/4.7.1.1 is also revised to include the algorithm used for determining the new allowable values.

*Date of issuance:* February 21, 2002.

*Effective date:* February 21, 2002.

*Amendment No.:* 155.

*Facility Operating License No. NPF–12:* Amendment revises the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** November 14, 2001 (66 FR 57125).

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

No significant hazards consideration comments received: No.

*South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina*

*Date of application for amendment:* August 20, 2001.

*Brief description of amendment:* This amendment revises the Technical Specifications by adding a footnote to Table 3.3–3 regarding the Steam Line Isolation and Engineered Safety Feature Actuation System (ESFAS) functions. This revision will allow V.C. Summer to exclude ESFAS steam line isolation instrumentation operability in Mode 3 when the main steam isolation valves, along with associated bypass valves, are closed and disabled, and eases the restriction of Specification 3.0.4 when performing reactor coolant system resistance temperature device cross calibrations at temperatures below the ESFAS P–12 Interlock for Low-Low $T_{avg.}$

*Date of issuance:* March 5, 2002.

*Effective date:* March 5, 2002.

*Amendment No.:* 156.

*Facility Operating License No. NPF–12:* Amendment revises the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** December 12, 2001 (66 FR 64301). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 5, 2002.

No significant hazards consideration comments received: No.

*Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California*

*Date of application for amendments:* December 11, 2001.

*Brief description of amendments:* The amendments revise Technical Specification (TS) 3.8.3, "Diesel Fuel Oil, Lube Oil, and Starting Air," to support the use of California Diesel fuel rather than the existing Environmental Protection Agency Clear diesel fuel, and reflect a change in the diesel generator load profile in Modes 1 through 4.

*Date of issuance:* March 5, 2002.

*Effective date:* March 5, 2002, to be implemented within 30 days of issuance.

*Amendment Nos.:* Unit 2—183; Unit 3—174.

*Facility Operating License Nos. NPF–10 and NPF–15:* The amendments revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** January 8, 2002 (67 FR 932). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 5, 2002.

No significant hazards consideration comments received: No.

*Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50–321, Edwin I. Hatch Nuclear Plant, Unit 1, Appling County, Georgia*

*Date of application for amendment:* August 31, 2001, supplemented January 24, 2002.

*Brief description of amendment:* The amendment revised the Technical Specifications to allow a one-time deferral of the Type A Containment Integrated Leak Rate test based on the risk-informed guidance in Regulatory Guide 1.174.

*Date of issuance:* February 20, 2002.

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

*Amendment No.:* 226.

*Facility Operating License No. DPR–57:* Amendment revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** October 17, 2001 (66 FR 52802). The supplement dated January 24, 2002, provided clarifying information that did not change the scope of the August 31, 2001, 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 February 20, 2002.

No significant hazards consideration comments received: No.

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

*Date of application for amendments:* August 31, 2001, supplemented November 15, 2001, and February 21, 2002.

*Brief description of amendments:* The amendments revised the Technical Specifications on a one-time basis to extend from 7 days to 14 days the completion time for the required actions associated with restoration of the 1B emergency diesel generator (EDG). The NRC review of the August 31, 2001, amendment request to extend the completion times for all of the EDGs to

14 days on a permanent basis is ongoing.

*Date of issuance:* February 22, 2002.

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

*Amendment Nos.:* 227/169.

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

*Date of initial notice in* ***Federal Register:*** October 17, 2001 (66 FR 52803). The supplemental letters dated November 15, 2001, and February 21, 2002, provide clarifying information that did not change the scope of the August 31, 2001, 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 February 22, 2002.

No significant hazards consideration comments received: No.

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

*Date of amendments request:* December 14, 2001.

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

*Date of issuance:* March 6, 2002.

*Effective date:* As of the date of issuance and shall be implemented by August 1, 2002.

*Amendment Nos.:* 153/145.

*Facility Operating License Nos. NPF–2 and NPF–8:* Amendments revise the Technical Specifications and associated Bases.

*Date of initial notice in* ***Federal Register:*** January 22, 2002 (67 FR 2928). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 6, 2002.

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:* August 10, 2001, as supplemented February 11, 2002.

*Description of amendment request:* The amendments revised Technical Specification 3.9.1, "Refueling Equipment Interlocks," to allow in-vessel fuel movement to continue with inoperable refueling equipment interlocks, provided (1) control rod withdrawals are blocked and (2) all control rods are verified to be inserted.

*Date of issuance:* March 6, 2002.

*Effective date:* March 6, 2002.

*Amendment Nos.:* 242, 274, and 232.

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

*Date of initial notice in* ***Federal Register:*** November 14, 2001 (66 FR 57126). The February 11, 2002, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 6, 2002.

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* April 20, 2001, as supplemented October 29, 2001, and November 14, 2001.

*Brief description of amendment:* Amends the Final Safety Analysis Report by changing the spent fuel pool (SFP) cooling analysis methodology to increase the evaluated heat removal capacity of the SFP cooling system.

*Date of issuance:* February 21, 2002.

*Effective date:* This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

*Amendment No.:* 37.

*Facility Operating License No. NPF–90:* Amendment does not revise the operating license or its appendices.

*Date of initial notice in* ***Federal Register:*** December 17, 2001 (66 FR 64998). The supplemental letters provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 21, 2002.

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* April 10, 2000, as supplemented by letters dated September 18, 2000, August 22, 2001, November 8, 2001, and January 15, 2002.

*Brief description of amendment:* The amendment incorporates new requirements into the Technical Specifications (TS) associated with steam generator (SG) tube inspection and repair, establishing an alternate voltage-based SG tube repair criteria.

*Date of issuance:* February 26, 2002.

*Effective date:* As of the date of issuance and shall be implemented prior to startup following the Cycle 4 refueling outage.

*Amendment No.:* 38.

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

*Date of initial notice in* ***Federal Register:*** May 31, 2000 (65 FR 34751). The supplemental letters provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 26, 2002.

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:* December 18, 2001.

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

*Date of issuance:* February 22, 2002.

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

*Amendment Nos.:* 92 and 92.

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

*Date of initial notice in* ***Federal Register:*** January 22, 2002 (67 FR 2931). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 22, 2002.

No significant hazards consideration comments received: No.

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

*Date of amendment request:* December 11, 2001.

*Brief description of amendment:* The amendment revises Surveillance Requirement (SR) 3.0.3 to extend the delay period before entering a limiting condition for operation following a missed SR from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement is added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

*Date of issuance:* March 4, 2002.

*Effective date:* March 4, 2002, and shall be implemented within 60 days from the date of issuance.

*Amendment No.:* 143.

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

*Date of initial notice in* ***Federal Register:*** January 8, 2002 (67 FR 935). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 4, 2002.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 11th day of March 2002.

For the Nuclear Regulatory Commission.

**John A. Zwolinski,**

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

[FR Doc. 02–6230 Filed 3–18–02; 8:45 am]

**BILLING CODE 7590–01–P**

## POSTAL RATE COMMISSION

### Briefing on Regulatory Developments

**AGENCY:** Postal Rate Commission.

**ACTION:** Notice of regulatory briefing.

**SUMMARY:** A delegation from Britain's Postal Services Commission (Postcomm), the independent regulator of Consignia (formerly the British Post Office), will present a briefing on Wednesday, March 27, 2002, beginning at 10 a.m., in the Postal Rate Commission's hearing room. The topic is recent regulatory developments in the United Kingdom. The briefing is open to the public.

**DATES:** March 27, 2002, 10 a.m.

**ADDRESSES:** Postal Rate Commission (hearing room), 1333 H Street NW., Washington, DC 20268–0001, suite 300.

**FOR FURTHER INFORMATION CONTACT:** Stephen L. Sharfman, general counsel, Postal Rate Commission, 202–789–6820.

**Steven W. Williams,**

*Secretary.*

[FR Doc. 02–6534 Filed 3–18–02; 8:45 am]

**BILLING CODE 7710–FW–M**

## SECURITIES AND EXCHANGE COMMISSION

**[Release No. 35–27497]**

### Filings Under the Public Utility Holding Company Act of 1935, as Amended (''Act'')

March 12, 2002.

Notice is hereby given that the following filing(s) has/have been made with the Commission pursuant to provisions of the Act and rules promulgated under the Act. All interested persons are referred to the application(s) and/or declaration(s) for complete statements of the proposed transaction(s) summarized below. The application(s) and/or declaration(s) and any amendment(s) is/are available for public inspection through the Commission's Branch of Public Reference.

Interested persons wishing to comment or request a hearing on the application(s) and/or declaration(s) should submit their views in writing by April 8, 2002, to the Secretary, Securities and Exchange Commission, Washington, DC 20549–0609, and serve a copy on the relevant applicant(s) and/or declarant(s) at the address(es) specified below. Proof of service (by affidavit or, in the case of an attorney at law, by certificate) should be filed with the request. Any request for hearing should identify specifically the issues of facts or law that are disputed. A person who so requests will be notified of any hearing, if ordered, and will receive a copy of any notice or order issued in the matter. After April 8, 2002, the application(s) and/or declaration(s), as filed or as amended, may be granted and/or permitted to become effective.

*E.ON AG, et al. (70–9985)*

E.ON AG (''E.ON''), a German company; E.ON's subsidiary companies, E.ON UK Verwaltungs GmbH (''E.ON UK''), E.ON UK plc, E.ON US Verwaltungs GmbH (''E.ON US''), E.ON Holdco (if formed) all located at E.ON-Platz 140479, Düsseldorf, Germany; Fidelia, Inc. (''Fidelia''), a finance company subsidiary organized in Delaware; E.ON North America Inc. (''E.ON NA''); Powergen plc (''Powergen''), a U.K. registered holding company; Powergen's direct and indirect wholly owned registered holding company subsidiaries, Powergen US Holdings Limited (''Powergen US Holdings''), Powergen US Investments, Powergen Luxembourg sarl, Powergen Luxembourg Holdings sarl, Powergen Luxembourg Investments sarl, Powergen US Investments Corp.(''PUSIC'' and together, ''Powergen Intermediate Companies''); Powergen US Funding LLC (''Powergen US Funding''), a financing vehicle for Powergen US Holdings, all located at 53 New Broad Street, London EC2M 1SL, United Kingdom; LG&E Energy Corp. (''LG&E Energy''), a Kentucky holding company exempt from registration under section 3(a)(1) of the Act, located at 220 West Main Street, Louisville, Kentucky 40232; LG&E Energy's utility subsidiaries Louisville Gas and Electric Company (''LG&E'') and Kentucky Utilities Company (''KU'' and together, ''Utility Subsidiaries''), One Quality Street, Lexington, Kentucky 40507; and LG&E Energy's nonutility companies located at 220 West Main Street, Louisville, Kentucky 40232 (''LG&E Nonutilities,'' together with LG&E Energy and the Utility Subsidiaries, ''LG&E Energy Group'' and collectively, ''Applicants'') have filed an application (''Application'') under sections 6(a), 7, 9(a), 10, 12, 13 of the Act and rules 45, 46, 52, 53, 54, 90 and 91 under the Act. Applicants request authority for various financing transactions and service agreements related to the acquisition by E.ON of Powergen and its subsidiaries (''Acquisition''). The Commission published a notice describing the application for the Acquisition (''Acquisition Application'') on December 21, 2001.[1] Following the Acquisition, E.ON intends to register under section 5 of the Act. Applicants intend that the LG&E Energy Group be transferred from the Powergen intermediate holding companies (''Powergen Intermediate Holding Companies'') and held indirectly by

[1] *See* E.ON AG plc, et al. HCAR No. 27482 (December 21, 2001).