For the Nuclear Regulatory Commission.

Joel T. Munday,

Acting Chief, Section 1, Project Directorate I, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 02–2738 Filed 2–4–02; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Weeks of February 4, 11, 18, 25, March 4, 11, 2002.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of February 4, 2002

Wednesday, February 6, 2002

9:25 a.m.
Affirmation Session (Public Meeting) (If needed)

9:30 a.m.
Briefing on Equal Employment Opportunity (EEO) Program (Public Meeting) (Contact: Irene Little, 301–415–7380)

Week of February 11, 2002—Tentative

There are no meetings scheduled for the Week of February 11, 2002.

Week of February 18, 2002—Tentative

Tuesday, February 19, 2002

1:55 p.m.
Affirmation Session (Public Meeting) (If needed)

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

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

Week of February 25, 2002—Tentative

Friday, March 1, 2002

9:30 a.m.
Briefing on Status of Office of the Chief Financial Officer (OCFO) Programs, Performance, and Plans (Public Meeting) (Contact: Lars Solander, 301–415–6080)

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

Week of March 4, 2002—Tentative

Monday, March 4, 2002

2 p.m.
Briefing on Status of Nuclear Waste Safety (Public Meeting) (Contact: Claudia Seelig, 301–415–7243)

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

Week of March 11, 2002—Tentative

There are no meetings scheduled for the Week of March 11, 2002.

* The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301 415–1292. Contact person for more information: David Louis Gamberoni (301) 415–1651.

Additional Information

By a vote of 5–0 on January 29 and 30, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Affirmation of 1) Dominion Nuclear Connecticut Inc. (Millstone Nuclear Power Station, Units 2 and 3) Petition for Reconsideration of CLI–01–24 and 2) Duke Cogema Stone & Webster (Savannah River Mixed Oxide Fuel Fabrication Facility); Georginas Against Nuclear Energy's Motion for Reconsideration of CLI–01–28'' be held on January 30, and on less than one week's notice to the public.

The NRC Commission Meeting Schedule can be found on the Internet at: *www.nrc.gov*

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, D.C. 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: January 31, 2002.

David Louis Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 02–2801 Filed 2–1–02; 10:23 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from January 11, 2002 through January 24, 2002. The last biweekly notice was published on January 22, 2002 (67 FR 2917).

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 March 7, 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.*

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

Date of amendments request: December 13, 2001.

Description of amendments request: The amendments would lower the maximum allowable differential pressure across the Engineered Safety Features (ESF) ventilation system units when tested at specified system flowrates.

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.

Technical Specification (TS) 5.5.11, Ventilation Filter Testing Program (VFTP) establishes a program for requiring testing of Engineered Safety Feature (ESF) filter ventilation systems in accordance with appropriate regulatory guidance.

PVNGS [Palo Verde Nuclear Generating Station] calculations 13–MC–HJ–0804 and 13–MC–HF–0902 were developed to document the design basis and testing standard positions that PVNGS has taken concerning the Control Room Essential Filtration System (CREFS) air filtration units (AFUs) and the ESF Pump Room Exhaust Air Cleanup System (PREACS) AFUs. These calculations established a lower design dirty filter differential pressure (D/P) to ensure that the AFUs are capable of delivering the design flows at 100% maximum dirty filter condition and also able to meet the adsorber residence time when the filters are clean. Design margin of the AFUs is validated via analyses performed in the referenced calculations and confirmed by the various startup and surveillance tests.

The analyses established a more restrictive design criteria than that which is currently listed in TS 5.5.11.d. The new D/P limit for the CREFS AFUs is less than or equal to 4.8 inches water gauge (iwg). The new D/P limit for the PREACS AFUs is less than or equal to 5.2 iwg. This applies to all three of the PVNGS units. Each PVNGS unit is equipped with two CREFS and two PREACS AFUs.

These essential AFUs are not event initiators. The essential CREFS and PREACS AFUs are used to mitigate the consequences of a postulated accident as discussed in Updated Final Safety Analysis Report (UFSAR) Sections 15.6 and 15.7. The proposed change in filter D/P for dirty filter conditions does not increase the probability of an accident previously evaluated.

The accident analyses that could be affected by the proposed changes to the CREFS and PREACS AFUs are addressed in the calculations which determine the expected radiological doses in the control room, at the Exclusion Area Boundary (EAB), and in the Low Population Zone (LPZ) resulting from postulated accidents. The efficiency of the essential AFU filter and charcoal adsorber as well as adsorber residence time and airflow rate are required parameters to evaluate the removal of radioactive gases and particulates from the postulated accidents evaluated in UFSAR Chapter 15. However, the proposed changes to the essential AFUs D/P limits ensure that PVNGS remains within existing licensing bases for radiological consequences of fuel handling accidents and LOCA events.

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

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

The purpose of the essential AFUs (CREFS and PREACS) is to mitigate the consequences of an accident and as such, they are not plant accident initiators.

The proposed changes in filter D/P limits for these essential AFUs do 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 operations. The proposed changes in the filter D/P limit for dirty filter conditions ensure that PVNGS remains within existing licensing bases for radiological consequences of fuel handling accidents and LOCA [loss-of-coolant accident] events and are not initiators of any new or different kinds of accidents.

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

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

The proposed change in the allowed maximum D/P across the filter in a dirty condition is a more conservative and restrictive change (less than or equal to 4.8 inches of water (iwg) for the CREFS units and 5.2 iwg for the PREACS units) than the current value of ''less than 8.4 iwg'' in Technical Specification 5.5.11.d. Under these conditions, the AFUs are required to deliver the design flows at a lower maximum D/P, which increases the structural safety margin of the filters. At the same time, the charcoal adsorber residence time requirements are met for the higher fan flowrate achieved with clean filters. The variations in diesel generator output voltage and frequency and its effects on the airflows and adsorber residence time are bounded by the design value parameters as demonstrated in calculations 13–MC–HJ–0804 and 13–MC–HF–0902. As such, the proposed changes ensure that PVNGS remains within existing licensing bases.

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

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

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

NRC Section Chief: Stephen Dembek.

Carolina Power & Light Company (CP&L), Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: December 20, 2001.

Description of amendment request: The amendment would revise Technical Specifications Section 5.6.5, ''Core Operating Limits Report (COLR)'' to add a report to the list of documents describing the approved methodologies.

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

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

* * * [The report proposed to be added to the COLR references is under generic review by NRC and, if approved, will be adopted for use.] Analyzed events are assumed to be initiated by the failure of plant structures, systems, or components. The core operating limits developed in accordance with the new methodology will be bounded by any limitations in the NRC acceptance in its safety evaluations of the new methodologies. The topical report associated with the new methodology demonstrates that the integrity of the fuel will be maintained during normal operations and that design requirements will continue to be met. The proposed change does not involve physical changes to any plant structure, system, or component. Therefore, the probability of occurrence for a previously analyzed accident is not significantly increased.

The consequences of a previously analyzed accident are dependent on the initial conditions assumed for the analysis, the behavior of the fuel during the analyzed accident, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. The proposed methodology continues to meet applicable design and safety analyses acceptance criteria. The proposed change does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. As a result, no analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident. The proposed change does not affect setpoints that initiate protective or mitigative actions. The proposed change ensures that plant structures, systems, or components are maintained consistent with the safety analysis and licensing bases. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

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

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

The proposed change does not involve any physical alteration of plant systems, structures, or components, other than allowing for fuel design in accordance with NRC approved methodologies. The proposed methodology continues to meet applicable criteria for LBLOCA [large-break loss-of-coolant accident] analysis. No new or different equipment is being installed. No installed equipment is being operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints that initiate protective or mitigative actions. As a result no new failure modes are being introduced. There are no changes in the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety is established through the design of the plant structures, systems, and components, through the parameters within which the plant is operated, through the establishment of the setpoints for the actuation of equipment relied upon to respond to an event, and through margins contained within the safety analyses. The proposed change in the methodology used for LBLOCA analyses does not impact the condition or performance of structures, systems, setpoints, and components relied upon for accident mitigation. The proposed change does not significantly impact any safety analysis assumptions or results. Therefore, the proposed change does not result in a significant reduction in the margin of safety.

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

Attorney for licensee: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: December 6, 2001.

Description of amendment request: The proposed amendments would revise the Technical Specification (TS) 3.7.16, ''Control Room Area Cooling System (CRACS),'' which currently requires entry into TS 3.0.3 when two trains of CRACS are inoperable. The proposed amendments would allow 6 hours to restore the operability of one train.

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

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

Response: No.

Loss of CRACS for the duration of the Completion Time is not a safety concern because equipment in the control area is suitable for considerably higher temperatures than will be experienced within the Completion Time.

The accidents evaluated in the UFSAR [Updated Final Safety Analysis Report] are not initiated by the CRACS or loss of the CRACS. Furthermore, the CRACS is not directly credited for mitigation of the accidents evaluated in the UFSAR. The CRACS does perform a support function to maintain environmental conditions for equipment that does help mitigate accidents. The proposed change does extend the total time from loss of a second required train until entry into the required MODEs. However, analysis confirms that the CRACS function is not required for a number of hours (i.e. 18 or more), which is substantially greater than the proposed Completion Time of 6 hours. The proposed Completion Time of 6 hours allows reasonable time for restoration prior to initiation of shutdown while leaving sufficient time to reach hot shutdown. The probability of an accident or event occurring during this Completion Time is acceptably low.

The current TS may require simultaneous reduction in power and shutdown of all three Units. Such action is not without some risk. Allowing the requested limited additional time to restore control area cooling reduces some risk factors by not changing plant power level in response to a minor problem that does not constitute a safety concern. If the initiation of shutdown of the affected units does become necessary, this change would allow operators more flexibility to sequence the shutdowns to minimize overall operator burden and the impact of simultaneous shutdowns.

In summary, this change will not involve a significant increase in the probability or consequences of any previously evaluated accident.

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

Response: No.

No new or different kind of accident has been identified as a result of this Technical Specification change.

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

Response: No.

The accidents evaluated in the UFSAR are not initiated by the CRACS or loss of the CRACS. The loss of the CRACS was screened out of the Oconee PRA and is not modeled in the present Oconee PRA as either an initiating event or as a support system failure. Temperature transient analyses calculate the time to reach the limiting design temperature of required systems, structures, or components supported by CRACS. Current analyses show CRACS is not required to perform a support function for at least 18 hours.

This 18 hour time is not used to calculate the consequences or impact on fission product barriers if CRACS is not restored. Instead this time is used to prioritize activities to restore CRACS and is substantially greater than the proposed 6 hour Completion Time. As discussed above, this allows reasonable time for restoration prior to initiation of shutdown, while leaving sufficient time to reach hot shutdown. Since either the CRACS function will be restored or the affected unit(s) will be shutdown, this change would not result in a change of, or challenge to, the design basis limit for a fission product barrier.

This change does not involve a departure from a method of evaluation used for evaluating behavior or response of the facility or supported components.

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

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

NRC Section Chief: Richard J. Laufer, Acting.

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

Date of amendment request: December 20, 2001.

Description of amendment request: The proposed amendments would revise the Technical Specification 5.6.5.b to eliminate the revision number and dates of the topical reports that contain the analytical methods used to determine the core operating limits. This proposed change is consistent with

TSTF (Technical Specification Task Force)–363.

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 implementation of the changes proposed in this LAR [license amendment request] involve a significant increase in the probability or consequences of an accident previously evaluated?

No. This LAR makes an administrative change to the Technical Specifications made necessary as part of Duke's implementation of revised NRC regulations. The changes proposed to these TS have no substantive impact on the Oconee licensing bases, nor Duke's ability to conservatively evaluate changes to these licensing bases. Therefore, the proposed changes have no impact on any accident probabilities or consequences.

2. Would implementation of the changes proposed in this LAR create the possibility of a new or different kind of accident from any accident previously evaluated?

No. This LAR makes administrative changes that have no impact on any accident analyses.

3. Would implementation of the changes proposed in this LAR involve a significant reduction in a margin of safety?

No. The proposed changes are administrative, an implementation of the revised 10CFR50.59 regulation. Implementation of the revised 10CFR50.59 regulation provides the necessary regulatory requirements to ensure that nuclear plants' margin of safety is preserved.

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

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

NRC Section Chief: Richard J. Laufer, Acting.

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

Date of amendment request: December 3, 2001.

Description of amendment request: Energy Northwest is requesting a revision to the technical specifications (TSs) and licensing and design bases to reflect the application of alternative source term methodology. The alternative source term analyses have been performed without crediting secondary containment during fuel handling accidents. As such, the proposed license amendment relaxes operability requirements during fuel handling and core alterations for: (1) secondary containment; (2) secondary containment isolation instrumentation; and (3) the standby gas treatment system. The alternative source term analyses have also been performed without crediting the main steam leakage control system; therefore, the licensing basis and the TS are being revised to reflect the proposed deactivation of the system. The license amendment request also addresses the establishment of secondary containment vacuum under adverse environmental conditions. In addition, the amendment request increases the allowed amount of unfiltered control room leakage into the control room.

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 alternative source term does not affect the design or operation of the facility; rather, once the occurrence of an accident has been postulated, the new source term is an input to evaluate the consequence. The implementation of the alternative source term methodology has been evaluated in revisions to the analyses of the following limiting design basis accidents at Columbia Generating Station:

- Control Rod Drop Accident
- Fuel Handling Accident
- Main Steam Line Break Accident
- Loss of Coolant Accident

Based upon the results of these analyses, it has been demonstrated that, with the requested changes, the dose consequences of these limiting events are within the regulatory guidance provided by the NRC for use with the alternative source term. This guidance is presented in 10 CFR 50.67 and associated Regulatory Guide 1.183, and Standard Review Plan Section 15.0.1.

Requirements for secondary containment operability, secondary containment isolation valves, and the standby gas treatment system during fuel movement or core alterations are being eliminated. This is acceptable because, with the application of alternative source term methodology, secondary containment is not credited for the fuel handling accident. The licensing basis is being revised to reflect the proposed deactivation of the main steam leakage control system. This is acceptable because, with the application of alternative source term methodology, no credit is assumed for the system in the accident analyses.

With regard to the Justification for Continued Operation regarding the establishment of secondary containment vacuum under adverse environmental conditions, the proposed changes to the secondary containment and standby gas treatment system Technical Specifications and application of alternative source term methodology ensures that secondary containment draw-down and bypass leakage are within the assumptions of the applicable safety analysis.

With regard to the previously-identified Unreviewed Safety Question pertaining to increased unfiltered control room in-leakage into the control room envelope, application of alternative source term methodology has shown that in-leakage rates in excess of tested values would result in control room doses below the regulatory limit.

Therefore, operation of Columbia Generating Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The alternative source term does not affect the design, functional performance or operation of the facility. Similarly, it does not affect the design or operation of any structures, systems or components equipment or systems involved in the mitigation of any accidents, nor does it affect the design or operation of any component in the facility such that new equipment failure modes are created.

Requirements for the main steam leakage control system are being deleted by this proposed amendment request. This is acceptable because the system no longer meets the criteria of 10 CFR 50.36. With the application of alternative source term methodology, no credit is assumed for the system in the accident analyses. Furthermore, since the main steam leakage control system is a mitigating system, it cannot create the possibility of an accident.

Requirements for secondary containment operability, secondary containment isolation valves, and the standby gas treatment system during fuel movement or core alterations are being eliminated. This is also acceptable because, with the application of alternative source term methodology, secondary containment is not credited for the fuel handling accident.

Therefore, the operation of Columbia Generating 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.

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

The changes proposed are associated with the implementation of a new licensing basis for Columbia Generating Station. Approval of the basis change from the original source term developed in accordance with TID–14844 to a new alternative source term as described in Regulatory Guide 1.183 is requested by this submittal. The results of the accident analyses revised in support of this submittal, and the requested Technical Specification changes, are subject to revised acceptance criteria. These analyses have been performed using conservative methodologies.

Safety margins and analytical conservatisms have been evaluated and are satisfied. The analyzed events have been carefully selected and margin has been retained to ensure that the analyses adequately bound postulated event scenarios. The dose consequences of these limiting events are within the acceptance criteria also found in the latest regulatory guidance. This guidance is presented in 10 CFR 50.67 and associated Regulatory Guide 1.183.

The proposed changes can be made while still satisfying regulatory requirements and review criteria, with significant margin. The changes continue to ensure that the doses at the exclusion area and low population zone boundaries, as well as the control room, are within the corresponding regulatory limits.

Therefore, operation of Columbia Generating Station in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: September 7, 2001 as revised December 17, 2001. This notice supersedes (66 FR 52799) published on October 17, 2001, which was based upon the licensee's application dated September 7, 2001.

Description of amendment request: The amendment would revise the Post Accident Monitoring Instrumentation Technical Specifications to ensure that licensee commitments to Regulatory Guide 1.97 are properly reflected.

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

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

Response:

The proposed amendment involves rewording or reformatting of technical specification requirements regarding certain post accident monitoring instrumentation at Indian Point 3, to improve the usability of the specification. The proposed rewording of the required channels for core exit temperature adopts the wording from the Standard Technical Specifications, which is applicable to the Indian Point 3 design. New condition entry statements are added in Condition C as an alternate formatting method which replaces the existing approach of using notes in the instrumentation list in Table 3.3.3–1, for certain instrument channels. Similarly, combining two existing functions into one new function is an improved formatting method that eliminates the need for a note in the Table. None of these proposed changes affect the requirements established in the existing specification.

Post accident monitoring instrumentation is a tool used by plant operators to conduct diagnostic activities outlined in plant emergency operating procedures. The presence or absence of this instrumentation does not influence accident initiators for accidents previously analyzed. Also, this instrumentation is not credited to support automatic responses for accident mitigating systems or equipment. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response:

The proposed amendment involves rewording or reformatting of technical specification requirements to improve the usability of the specification for certain post accident monitoring instrumentation at Indian Point 3. The proposed amendment does not involve any changes to plant equipment, setpoints, or the way in which the plant is operated. The proposed amendment maintains the existing requirements for post accident monitoring instrumentation using an improved presentation format. Therefore the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response:

The proposed amendment involves rewording or reformatting of technical specification requirements to improve the usability of the specification for certain post accident monitoring instrumentation at Indian Point 3. The proposed rewording of the required channels for core exit temperature adopts the wording from the Standard Technical Specifications, which is applicable to Indian Point 3. Use of the standard wording ensures consistent application of the requirements for this post accident monitoring function. Similarly, reformatting the specification to use new condition entry statements, rather than the existing notations in the Table will improve the usability of the specification and ensure that the intended requirements will be consistently applied.

The proposed changes do not delete or modify existing requirements or add new requirements. The changes involve rewording or reformatting of existing requirements and provide an improved method of stating the requirements intended in the existing specification. Therefore, the proposed amendment 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. John Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Section Chief: Joel T. Munday, Acting.

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 amendment request: January 16, 2002.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) to permit functional testing of the emergency diesel generators (EDGs) to be performed during power operation. The proposed changes will add a footnote to Surveillance Requirement 4.8.1.1.2.g.7 regarding the 24-hour functional test of the EDGs. The changes are based on an integrated review of deterministic design basis factors, and an evaluation of plant risk using probabilistic safety assessment (PSA) techniques.

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

The function of the emergency diesel generators is to supply emergency power in the event of a LOOP. Operation of the EDGs is not a precursor to any accident. The EDGs provide assistance in accident mitigation. There are no technical changes related to the acceptance criteria of the surveillance requirement. The proposed change requesting that the scheduling aspects of the surveillance requirements be changed to accommodate improved planning capability for testing does not affect the accident analyses. The EDG that is being tested will be considered inoperable however, the remaining required EDGs would be operable during the test and they are capable of supporting the safe shutdown of the plant. The Probabilistic Safety Assessment (PSA) results fall below the Acceptance Guidelines for TS changes contained in Regulatory Guides 1.174 and 1.177; therefore, the risk of performing the EDG 24-hour run during POWER OPERATION has only a small quantitative impact on plant risk. Therefore, the proposed change to permit the 24-hour functional test of the EDGs to be performed during POWER OPERATION does not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed change does not include any physical changes to plant design or a change to current Surveillance Requirement acceptance criteria. Performance of the Surveillance Requirement during POWER OPERATION results in equipment out of service, inoperable EDG, which is addressed by current Technical Specification limiting condition for operation. Therefore, performance of the EDG 24-hour functional testing during POWER OPERATION does 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 amendment would not involve a significant reduction in a margin of safety.

The proposed changes are associated with surveillance requirements for the EDGs. The proposed changes allow the EDG 24-hour functional testing to be performed during POWER OPERATION. Performing the functional test during POWER OPERATION will not impact the plant design bases or safety analyses because the affected EDG will be declared inoperable during the test. During the time that the EDG in test is declared inoperable, the system is considered to be exempt from the single failure criterion such that adequate emergency power will remain available to support the system design bases.

From a design basis perspective, the inoperable EDG effectively represents a single failure for the system. Since the emergency power system is designed to accomplish its system safety functions with only two of the three EDGs in service, and recovery of a failed component is not credited in the plant safety analysis (i.e., the single failure remains in effect for the entire accident sequence), removing an EDG from service to perform a 24-hour functional test during POWER OPERATION will not reduce the margin of safety assumed in the plant safety analyses.

The Probabilistic Safety Assessment (PSA) results fall below the Acceptance Guidelines for TS changes contained in Regulatory Guides 1.174 and 1.177. Therefore, the risk of performing the EDG 24-hour run during POWER OPERATION has only a small quantitative impact on plant risk.

An integrated assessment of the risk impact of performing the 24-hour functional test during POWER OPERATION for a single inoperable EDG has determined that the risk contribution is small and is within regulatory guidelines. Therefore, facility operation in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of application for amendment: December 26, 2001.

Brief description of amendment: The licensee proposed to revise Table 3.6.1.3–1, ''Secondary Containment Bypass Leakage Paths Leakage Rate Limits,'' of the Technical Specifications to re-designate two feedwater system air-operated primary containment isolation valves (PCIVs) as simple check valves. Upon approval by the NRC staff, the licensee would modify the air-operated PCIVs to become simple check valves. The simple check valves will perform the same function as the air-operated valves during normal and accident conditions. This design change only affects the nonsafety-related remote testing and position indication design features of the feedwater check valves.

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

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

No. The proposed amendment does not affect the probability of previously evaluated accidents because the affected PCIVs were not presumed to be initiators or precursors of any accident. The modified valves will continue to perform the same function as before. The modified valves will not alter or prevent the ability of existing structures, systems, or components to perform their intended safety or accident-mitigating functions depicted in the Updated Final Safety Analysis Report. The proposed amendment and the underlying design change will not prevent the unit to continue to comply with applicable regulatory requirements. As a result, the proposed amendment will not alter the conditions or assumptions used in previously evaluated accidents, specifically, the feedwater line break accident outside containment, and the loss-of-coolant accident.

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

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

No. The proposed amendment would lead to modification of air-operated PCIVs to become simple check valves. The modified valves will continue to perform the same function (i.e., prevent back flow in the feedwater line). Furthermore, the modified valves would not alter or prevent the ability of structures, systems, or components to perform their intended safety or accident mitigating functions. Thus, previously evaluated accident scenarios would not be altered by the proposed amendment.

Accordingly, the proposed amendment and the resulting design modification do not create any new or different kind of accident from any accident previously evaluated.

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

No. The proposed amendment does not change any analysis methodology, safety limits or acceptance criteria. The modified valves will have the same level of performance as before.

Therefore, operation in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: L. Raghavan (Acting).

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

Date of amendment request: December 19, 2001.

Description of amendment request: The proposed amendment would extend the completion time under Technical Specification (TS) Section 3.8.4.A to allow replacement of 125 VDC Batteries 1D1 and 1D2 while at power (Mode 1). The proposed amendment would add required actions 3.8.4.A.2.1 and 3.8.4.A.2.2 as one-time-only alternates and a conditional note following 3.8.4.A.1 to allow replacement of the 125 VDC batteries during a 10-day period for each battery. This TS change would be applicable one-time only, for each battery.

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

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

During the replacement of the existing station batteries, a temporary battery will provide the same function as the battery being removed. Even though this temporary battery will not meet seismic requirements, it will be assembled from safety-related Class 1E cells. The temporary battery will be subjected to surveillance testing prior to being utilized to confirm serviceability. The respective DC bus will be continuously energized by the existing battery charger. A backup swing charger will also be available which is a normal part of system configuration.

This one-time change also requires that required features be declared inoperable when the associated 125 VDC source is inoperable and the redundant required feature(s) are also inoperable for at least four hours. This action is intended to provide assurance that a loss of onsite power, during the period that a 125 VDC source is inoperable, does not result in a complete loss of safety function of critical systems. The completion time is intended to allow the operator time to evaluate and repair any discovered inoperabilities.

Due to the limited duration of the activity, the very low probability of a seismic event over this limited extended completion time, and the planned implementing contingency actions, a significant increase in the probability of an accident previously evaluated does not occur. The proposed change does not affect accident initiators or precursors, or design assumptions for the systems or components used to mitigate the consequences of an accident as analyzed in Chapter 15 of the DAEC UFSAR. The other division of DC power will remain operable to support design mitigation capability. Therefore, the proposed one-time completion time TS amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

During the replacement of the existing station batteries, a temporary battery will provide the same function as the batteries being removed. Even though this temporary battery does not meet the seismic requirements, it possesses adequate capacity to fulfill the safety-related requirements of supplying necessary power to the associated 125VDC bus. Because the temporary battery will perform like the station battery that is currently installed, no new electrical or functional failure modes are created. The temporary battery will be located in the turbine building which is non-seismic. The temporary battery will not be placed into seismically mounted racks. Thus, a seismic failure of this temporary battery is possible. The failure, if it does occur, would not create a new or different kind of accident from accidents previously evaluated.

This one-time change also requires that required features be declared inoperable when the associated 125 VDC source is inoperable and the redundant required feature(s) are also inoperable for at least four hours. This action is intended to provide assurance that a loss of onsite power, during the period that a 125 VDC source is inoperable, does not result in a complete loss of safety function of critical systems. The completion time is intended to allow the operator time to evaluate and repair any discovered inoperabilities.

The proposed one-time change does not introduce any new accident initiators or precursors or any new design assumptions for those systems or components used to mitigate the consequences of an accident. Therefore, the possibility of a new or different kind of accident from any previously evaluated has not been created. Thus, the proposed one-time completion time extension TS amendment does not create the possibility of a new of different kind of accident from any previously evaluated.

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

During the replacement of the existing station batteries, a temporary safety-related battery will perform the same function as the battery being removed. Even though this battery will not be seismically mounted, it will be assembled from safety-related Class 1E cells. The battery is functionally similar to the safety-related battery that is already installed. It will possess adequate capacity to fulfill the requirements of the associated 125VDC bus. The proposed replacement activity will not prevent the plant from mitigating a Design Basis Accident (DBA) during events that result in the loss of power from the temporary battery. In these cases, the remaining DC power supporting the design mitigation capability will be maintained. Due to the limited duration of the activity, the very low probability of a seismic event over this limited extended completion time, and the planned implementing contingency actions, a significant reduction in the margin of safety will not result. The associated DC bus will always be supplied by either the temporary battery and/or the battery charger at all times. In addition a spare swing battery charger is available. As a result, there is no significant reduction in the margin of safety.

This one-time change also requires that required features be declared inoperable when the associated 125 VDC source is inoperable and the redundant required feature(s) are inoperable for at least four hours. This action is intended to provide assurance that a loss of onsite power, during the period that a 125 VDC source is inoperable, does not result in a complete loss of safety function of critical systems. The completion time is intended to allow the operator time to evaluate and repair any discovered inoperabilities.

Therefore, this proposed amendment 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: Al Gutterman, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

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

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

Date of amendment request: July 30, 2001, as supplemented September 7, October 16, and December 5, 2001, and January 18, 2002.

Description of amendment request: This notice supercedes a notice published on November 14, 2001 (66 FR 57123).

The proposed amendments would revise Technical Specification 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' to allow a one-time deferral of the Type A containment integrated leakage rate test (ILRT) at the Susquehanna Steam Electric Station (SSES), Units 1 and 2. The Unit 1 test would be deferred to no later than May 3, 2007, and the Unit 2 test would be deferred to no later than October 30, 2007, resulting in an extended interval of 15 years for performance of the next ILRT at each unit. Additionally the proposed amendments would allow a one-time deferral of the drywell-to-suppression chamber bypass leakage test, Surveillance Requirement (SR) 3.6.1.1.2, so that it would continue to be conducted along with the ILRT, consistent with current practice.

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?

The frequency of Type A testing does not change the probability of an event that results in core damage or vessel failure. Primary containment is the engineered feature that contains the energy and fission products from evaluated events. The SSES IPE [Individual Plant Examination] documents events that lead to containment failure. The frequency of events that lead to containment failure does not change because it is not a function of the Type A test interval. Containment failure is a function of loss of safety systems that shutdown the reactor, provide adequate core cooling, provide decay heat removal, and loss of drywell sprays.

Similarly, the frequency of the SR 3.6.1.1.2 bypass test does not change the probability of an event that results in core damage or vessel failure since they are not a function of the bypass test.

The consequences of the evaluated accidents are the amount of radioactivity that is released to secondary containment and subsequently to the public. Normally, extending a test interval increases the probability that a Structure, System, or Component will fail. However, NUREG–1493, Performance-Based Containment Leak-Test Program, states that calculated risks in BWR's is very insensitive to the assumed leakage rates. The remaining testing and inspection programs provide the same coverage as these tests, and will maintain containment leakage at appropriately low levels. Any leakage problems will be identified and repairs will be made. Additionally, the containment is continuously monitored during power operation. Anomalies are investigated and resolved. Thus there is a high confidence that [containment] integrity will be maintained independent of the Type A test and SR 3.6.1.1.2 bypass test frequency.

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

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

Primary containment is designed to contain energy and fission products during and after an event. The SSES IPE identifies events that lead to containment failure. The proposed revision to the Type A and SR 3.6.1.1.2 test interval does not change this list of events. There are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure mode creating an accident or affecting mitigation of an accident.

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

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

The proposed one time extension to the Type A test frequency and the frequency of SR 3.6.1.1.2 from 10 to 15 years does not involve a significant reduction in margin of safety.

The tests are performed to ensure the degree of reactor containment structural integrity and leak-tightness considered in the plant safety analysis is maintained. These proposed changes do not affect the degree of leak-tightness nor structural integrity of the containment. These proposed changes only affect the frequency by which the tests are performed. The test acceptance criteria are not affected.

The proposed TS changes do not involve a change in the manner in which any plant system is operated or controlled.

The proposed TS changes do not affect the availability of equipment associated with containment integrity that is assumed to operate in the plant safety analysis.

The NUREG–1493 generic study of the effects of extending containment leakage testing found that a 20-year interval in Type A leakage testing resulted in an imperceptible increase in risk to the public. PPL analyses determined the total integrated risk and [Large Early Release Frequency] LERF increase is not significant. NUREG–1493 found that, generically, the design containment leakage rate contributes a very small amount of individual risk and would have minimal affect since most potential leakage paths are detected by Type B and Type C testing. Type B and Type C testing combined with visual inspection programs will maintain containment leakage at appropriately low levels.

The vacuum breaker leakage test (SR 3.6.1.1.3) and stringent acceptance criteria, combined with the negligible non-vacuum breaker leakage area and thorough periodic visual inspection, provide an equivalent level of assurance as the SR 3.6.1.1.2 bypass test. PPL analyses determined the total integrated risk and LERF increase is not significant.

The combination of the factors described above ensures that the proposed changes do not represent a significant reduction on 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: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Section Chief: J. Munday, Acting.

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 amendment request: November 1, 2001.

Description of amendment request: The proposed changes would modify the provisions under which equipment may be considered operable when either its normal or emergency power source is inoperable. Technical Specifications (TSs) Section 3.0.5 will be deleted under this proposal, and additional limiting conditions for operation (LCO) will be incorporated into electrical power systems TS 3.8.1.1, A.C. Sources—Operating. The corresponding TS Bases will be modified accordingly. The proposed changes are consistent with the recommendations contained in NUREG–1431, Rev. 2, "Standard Technical Specifications for Westinghouse 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. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The design of the AC electrical power system ensures that sufficient power will be available for engineered safeguards equipment required for safe shutdown of the facility and mitigation of accident conditions. Initial conditions of design basis accidents and transients in the Accident Analysis assume required engineered safeguards systems are operable and will function in order to maintain plant response within design limits. The proposed changes to action times do not affect the probability that any accident will occur. Since the minimum configuration of equipment assumed in the Accident Analysis will remain available, there will similarly be no increase in consequences of any accident.

The proposed changes to action times are consistent with the Westinghouse Standard Technical Specification (STS) requirements. This specification is intended to provide assurance that an event coincident with a failure of the associated normal or emergency power supply will not result in complete loss of safety function of critical required systems. The completion time allows the operator time to evaluate and repair any discovered inoperability. The given time periods are considered acceptable because they minimize risk while allowing time for restoration before subjecting the unit to transients associated with shutdown. These completion times take into account the capacity and capability of the remaining AC sources, a reasonable time for repairs and the low probability of a design basis accident occurring during this period.

With failure of one offsite power source, the remaining operable offsite circuit and diesel generators (DG) are adequate to supply electrical power to the onsite Class 1E electrical distribution system. At least one complete train of equipment will continue to operate in the same manner as assumed in the analyses to mitigate a design basis accident, given a failure of one component in a redundant train.

With both required offsite circuits inoperable, onsite emergency AC sources remain available to maintain the unit in a safe shutdown condition in the event of a design basis accident (DBA) or transient. The action completion time is reduced to 12 hours in this case. At least one complete train of equipment will operate as assumed in the analyses to mitigate a design basis accident, given a failure of one component in a redundant train.

With a single emergency diesel generator inoperable, the remaining operable DG and offsite power circuits are adequate to supply power to the onsite Class 1E electrical distribution system. Required actions ensure that a loss of offsite power during this period does not result in a complete loss of safety functions. Four hours is considered an acceptable time period to minimize risk during this condition, while allowing reasonable time for repair.

In any of these scenarios at least one train of equipment will be available to mitigate an accident and bring the plant to a safe shutdown condition, as assumed in the Accident Analysis. There will be no impact to radiological dose consequences.

Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

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

Expanding the allowable out of service time consistent with requirements of Standard Technical Specifications does not introduce any new or different failure from any previously evaluated or change the manner in which safety systems are operated. The associated system and equipment configurations are no different from those previously evaluated. The change in allowable action times have been considered and determined to be acceptable, without causing additional risk. The conditions of TS 3.8.1 continue to ensure that an event coincident with a failure of the associated normal or emergency power supply will not result in complete loss of safety function of critical required systems.

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

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

The power sources and distribution systems are designed to ensure sufficient power is available to supply safety related equipment required for safe shutdown of the facility and mitigation and control of accident conditions. Operability requirements are consistent with initial conditions assumed in the accident analysis. The proposed changes continue to provide assurance that an event coincident with failure of an associated diesel generator or offsite power circuit will not result in complete loss of safety function of critical required redundant systems or equipment.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: August 7, 2001.

Description of amendment request: SCE&G proposes a change to Table 3.3–2 of the Virgil C. Summer Nuclear Station Technical Specifications Surveillance Requirements to include a response time requirement of 0.5 seconds for the Source Range (SR) Neutron Flux Reactor Trip. The proposed change results from SCE&G's review of Westinghouse Nuclear Safety Advisory Letter NSAL–00–016. This NSAL notified SCE&G that the SR Neutron Flux Reactor Trip is implicitly credited within the accident analyses for the Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from Subcritical event during Modes 3, 4, and 5.

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

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

This change enhances the operability requirements of the SR Neutron Flux Instrumentation (NI) system by requiring response time testing. The performance of the required response time testing for the SR Neutron Flux Channels does not contribute to the initiation of any accident previously evaluated. Testing will be done during normal channel calibration when the SR Reactor Trip function is not required to be operable. During and following the required response time testing, there will be no adverse affect on the design and operation of the NSSS, BOP, and fluid and auxiliary system which are important to safety. Since the reactor coolant pressure boundary integrity and normally operating systems are not adversely impacted, the probability of occurrence of an accident evaluated in the VCSNS FSAR is no greater than the original design basis of the plant.

The availability of a reactor trip on the SR trip function with a defined response time of 0.5 seconds ensures that the event consequences of a RWFS event in Modes 3, 4, or 5 remain bounded by the current FSAR analysis. This is accomplished by ensuring that the reactor is shutdown before any significant power is generated.

With this change, periodic time response testing of the SR reactor trip function will be required to demonstrate that SR reactor trip function can be completed within the time limit assumed in the accident analyses. This enhanced operability requirement of the SR NI system provides additional assurance that the plant will be operated within its design and licensing basis. Any event that requires the mitigative function of this system will remain bounded by the analysis documented in Chapter 15 of the FSAR. No adverse hardware, software, setpoint or procedure changes are associated with this change. Furthermore, during and following the required response time testing, there will be no adverse affect on the design and operation of the NSSS, BOP, and fluid and auxiliary systems which are important to safety. Given the above, there is no potential for additional releases as a result of this activity. Therefore, no increase in any previously evaluated accident consequences will occur.

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

Enhancing the operability requirement for a Reactor Protection System input can not be considered an accident precursor. This change adds response time testing to the SR NI system which assures that the accident analysis, including assumptions, is maintained. No hardware, software, operational practices or instrumentation setpoints are being revised. No change to plant operating characteristics or philosophy result from this change. Therefore, the possibility of an accident of a different type is not being created.

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

TS Table 3.3–2 currently states that the response time for the SR NI is not applicable. However, the inherent assumption that this system will be the principal system to mitigate the rod withdrawal from subcritical accident is described in FSAR 15.2.1. The margin of safety is enhanced by the addition of an administrative requirement, to assure the safety analysis assumptions are satisfied. The maximum response time of 0.5 seconds is consistent with the maximum for Power Range and is conservative enough to limit the potential excursion to a safe value prior to tripping the plant.

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

Attorney for licensee: Thomas G. Eppink, South Carolina Electric & Gas

Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: Richard Laufer, 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: January 9, 2002.

Description of amendment requests: The proposed amendments would revise Technical Specification 5.4, Technical Specifications (TS) Bases control. Specifically, TS 5.4.2 and TS 5.4.2.b would be revised to replace the word ''involve'' with ''require'' and delete the term ''unreviewed safety question,'' respectively. The proposed changes are pursuant to the revised regulations in Title 10, Code of Federal Regulations (10 CFR) Section 50.59 which eliminated the term ''unreviewed safety question.''

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 replaces the word ''involve'' with ''require'' and deletes reference to the term ''unreviewed safety question'' consistent with 10 CFR 50.59. Deletion of the term ''unreviewed safety question'' was approved by the Nuclear Regulatory Commission (NRC) with the revision to 10 CFR 50.59. Consequently, the probability of an accident previously evaluated is not significantly increased. Changes to the Technical Specification (TS) Bases are still evaluated in accordance with 10 CFR 50.59. As a result, the consequences of any accident previously evaluated are not significantly affected.

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

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

Response: No.

The proposed changes do 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 plant operation. These changes are considered administrative changes and do not modify, add, delete, or relocate any technical requirements in the TS.

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

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

Response: No.

The proposed changes will not reduce a margin of safety because they have no effect on any safety analyses assumptions. Changes to the TS Bases that result in meeting the criteria in paragraph (c)(2) of 10 CFR 50.59 will still require NRC approval. The proposed changes to TS 5.4.2 are considered administrative in nature based on the revision to 10 CFR 50.59.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment 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 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 amendment request: January 4, 2002.

Description of amendment request: The proposed amendment would change the Safety Limit Minimum Critical Power Ratio (SLMCPR) for single loop operation (SLO) in Technical Specification (TS) 2.1.1.2 to reflect the results of a cycle-specific calculation.

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

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

The derivation of the revised SLO [single loop operation] SLPCPR for [safety limit critical power ratio] Plant Hatch Unit 1 Cycle 21 for incorporation into the TS, and its use to determine cycle-specific thermal limits, has been performed using NRC-approved methods and procedures. The procedures incorporate cycle-specific parameters and reduced power distribution uncertainties in the determination of the value for the SLMCPR. These calculations do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient.

The basis of the MCPR Safety Limit is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. The new SLO SLMCPR preserves the existing margin to transition boiling and the probability of fuel damage is not increased. Therefore, the proposed change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed change is the result of a cycle-specific application of NRC-approved methods to the Unit 1 Cycle 21 core reload. This change does not involve any new method for operating the facility and does not involve any facility modifications. No new initiating events or transients result from this change. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety as defined in the TS bases will remain the same. Cycle-specific SLMCPRs are calculated using NRC-approved methods and procedures, and meet the current fuel design and licensing criteria. The SLO SLMCPR will be high enough to ensure that greater than 99.9% of all fuel rods in the core are expected to avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity. Therefore, the proposed TS change does not involve a reduction in the margin of safety.

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

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Richard J. Laufer, Acting.

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 amendment request: December 14, 2001.

Description of amendment request: A change is proposed to Surveillance Requirement (SR) 3.0.3 to allow a longer period of time to perform a missed surveillance. The time 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 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 December 14, 2001.

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

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Richard J. Laufer, Acting.

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: October 24, 2001.

Description of amendment request: The proposed amendment would relocate various Technical Specifications (TSs) to the Technical Requirements Manual (TRM). Their associated Bases will also be relocated to the TRM to be consistent with relocation of the various TSs. In addition, the proposed amendment corrects various typographical and page numbering errors, deletes an outdated one-time exception, and makes minor formal changes to improve consistency.

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

Response: No.

This request involves relocation of information to the Technical Requirements Manual and administrative changes only. No actual plant equipment or accident analyses will be affected by the proposed changes. Therefore, the proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

This request involves relocation of information to the Technical Requirements Manual and administrative changes only. The proposed change does not alter the performance of the equipment or the manner in which the equipment will be operated. The equipment will still be verified by test, if applicable, in accordance with applicable surveillance requirements. Changing the location of these requirements and surveillances from Technical Specifications to the Technical Requirements Manual will not create any new accident initiators or scenarios. Since the proposed changes only allow activities that are presently approved and conducted, no possibility exists for a new or different kind of accident from those previously evaluated.

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

Response: No.

This request involves relocation of information to the Technical Requirements Manual and administrative changes only. No actual plant equipment or accident analyses will be affected by the proposed change. Additionally, the proposed changes will not relax any criteria used to establish safety limits, will not relax any safety systems settings, or will not relax the bases for any limiting conditions of operation. Therefore, the proposed changes will not impact 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: A. H. Gutterman, Esq., Morgan, Lewis &

Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

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: November 5, 2001.

Description of amendment request: The proposed amendments would revise specific requirements of Technical Specification (TS) Section 6.0, ''Administrative Controls.'' The proposed amendments include relocating specific TS administrative control requirements to licensee-controlled documents; updating specific management titles to more generic title positions; updating requirements to be consistent with current industry standards; and reformatting, renumbering, and rewording existing requirements for better readability. The proposed changes include Items 1 thru 125, and 127 in Table 1 of Attachment 1 of the licensee's submittal.

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

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

The proposed changes involve reformatting, renumbering, and rewording of the existing TS. These modifications involve no technical changes to the existing TS. As such, these changes are administrative in nature and do not effect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes involve reformatting, renumbering, and rewording of the existing TS. The changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes involve reformatting, renumbering, and rewording of the existing TS. The changes are administrative in nature and will not involve any technical changes. The changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. Also, since these changes are administrative in nature, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

More Restrictive Changes

The proposed changes designated as ''More Restrictive'' (M) technical changes involve adding more restrictive requirements to the existing TS by either making current requirements more stringent or by adding new requirements that currently do not exist. These changes have been evaluated to not be detrimental to plant safety. These changes are modifications of requirements to provide consistency with the Improved Standard Technical Specifications recommended in NUREG–1431. The proposed changes include Items 39, 51, 129 and 130 in Table 1.

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

The proposed changes provide more stringent requirements for operation of the facility. The more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more stringent requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The imposition of more stringent requirements either has no impact on or increases the margin of plant safety. As noted in the discussion of the changes, each change in this category, by definition, provides additional restrictions to enhance plant safety. The changes maintain requirements within the safety analyses and licensing basis. Therefore, these changes do not involve a significant reduction in a margin of safety.

Less Restrictive Changes L.1

Current TS 6.8.3.i, ''Diesel Fuel Oil Testing Program,'' requires properties for ASTM 2D fuel oil to be within limits within 30 days following sampling. The proposed change will increase the time in which compliance must be verified following sampling from 30 days to 31 days. This change is reasonable based on the relatively small increase in time and the probability of a major problem being found that would prevent the diesel generator from starting and operating. The proposed change, Item 70 in Table 1, is consistent with NUREG–1431.

In accordance with the criteria set forth in 10 CFR 50.92, the South Texas Project has evaluated this proposed TS change and determined that it involves no significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change extends the allowed completion time from 30 days to 31 days to verify that diesel fuel sample properties comply with ASTM 2D. This change does not affect the probability of an accident. Diesel fuel oil is not an initiator of any analyzed event. The consequences of an accident are not increased significantly because of the remote probability of an event occurring during the 24-hour period. Also, the probability of a major problem being found which would prevent the diesel generator from starting and operating is remote. The change will not alter the ability to mitigate an accident or transient event. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The change will not impose any new or different requirements or eliminate any existing requirements. 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 change involve a significant reduction in a margin of safety?

The proposed change extends the allowed completion time from 30 days to 31 days to verify that diesel fuel sample properties comply with ASTM 2D. The change does not significantly decrease the margin of safety because of the remote probability of an event occurring during the 24-hour period. Also, the probability of a major problem being found which would prevent the diesel generator from starting and operating is remote. The safety analysis assumptions will still be maintained. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Less Restrictive Change L.2

Current TS 6.9.1.2 and 6.9.1.2.a require annual submittal of an Occupational Radiation Exposure Report by March 1 of the calendar year following the exposures. The submittal date is revised to April 30. This change is consistent with previous comprehensive revisions to 10 CFR Part 20. The report is provided to supplement the information required by 10 CFR 20.2206(b), which is filed on or before April 30 in accordance with 10 CFR 20.2206(c). The proposed change, Item 76 in Table 1, is consistent with NUREG–1431.

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

The proposed change does not result in any changes in hardware or methods of operation. The change in date for submittal of ''after the fact'' information is not considered in the safety analysis and cannot initiate or affect the mitigation of an accident in any way. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The change will impact only the administrative requirements for submittal of information and does not directly impact the operation of the plant. 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 change involve a significant reduction in a margin of safety?

The proposed change does not impact the margin of safety since the margin of safety is not dependent on the submittal of information. The safety analysis assumptions will still be maintained. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Less Restrictive Change L.3

Current TS 6.9.1.3 requires annual submittal of a Radiological Environmental Operating Report by May 1 of each year. The submittal date is revised to May 15. This is an interval increase of 15 days. There is no requirement for the NRC to approve this report and 10 CFR [Part] 50 does not specify a specific reporting date. The proposed change, Item 82 in Table 1, is consistent with NUREG–1431.

In accordance with the criteria set forth in 10 CFR 50.92, the South Texas Project has evaluated this proposed TS change and determined that it involves no significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change does not result in any changes in hardware or methods of operation. The change in date for submittal of ''after the fact'' information is not considered in the safety analysis and cannot initiate or affect the mitigation of an accident in any way. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The change will impact only the administrative requirements for submittal of information and does not directly impact the operation of the plant. 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 change involve a significant reduction in a margin of safety?

The proposed change does not impact the margin of safety since the margin of safety is not dependent on the submittal of information. The safety analysis assumptions will still be maintained. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Less Restrictive Change L.4

Current TS 6.9.1.4 requires annual submittal of a Radioactive Effluent Release Report within 60 days after January 1 of each year. The submittal date is revised to May 1. This is an interval increase of approximately 60 days. The proposed change, Item 85 in Table 1, is consistent with NUREG–1431.

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

The proposed change does not result in any changes in hardware or methods of operation. The change in date for submittal of ''after the fact'' information is not considered in the safety analysis and cannot initiate or affect the mitigation of an accident in any way. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The change will impact only the administrative requirements for submittal of information and does not directly impact the operation of the plant. 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 change involve a significant reduction in a margin of safety?

The proposed change does not impact the margin of safety since the margin of safety is not dependent on the submittal of information. The safety analysis assumptions will still be maintained. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Less Restrictive Change L.5

The details specifying responsibility for initiating the Radiation Work Permit (RWP) surveillance frequency are being deleted. The requirement of current TS 6.12.1.c pertains to the individual qualified in radiation protection responsible for providing control over the activities in a high radiation area, including the performance of periodic radiation surveillances. The details specifying responsibility for the surveillance frequency in the RWP have no bearing on the requirements for entering a high radiation area. RWP details are controlled by plant procedures. Deleting these details eliminates ambiguity in the TS and the possibility for a misinterpretation of the TS requirements. The proposed change is provided in Table 1 as Item 103.

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

The proposed change eliminates ambiguity in the TS details specifying responsibility for the surveillance frequency in the Radiation Work Permit. The proposed change does not result in any changes in hardware or methods of operation. The details pertaining to the surveillance frequency in the Radiation Work Permit are not considered in the safety analysis and cannot initiate or affect the mitigation of an accident in any way. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed [change] does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The change will not impose any new or different requirements or eliminate any existing requirements. 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 change involve a significant reduction in a margin of safety?

The proposed change does not impact the margin of safety since the margin of safety is not dependent on who initiates the surveillance frequency of the Radiation Work Permit. The safety analysis assumptions will still be maintained. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Less Restrictive Change L.6

The details specifying the individuals responsible for performance of the review of the use of overtime are being deleted, and the frequency at which the overtime review is performed is being changed from monthly to periodic. The details specifying responsibility for performance of the overtime review and the frequency of review are controlled by plant procedures. The proposed changes are consistent with the programmatic controls required by NUREG–1431. The proposed changes are provided in Table 1 as Item 30a.

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

The proposed changes delete the details specifying the individuals responsible for performance of the overtime use review, and changes the frequency at which the overtime review is performed from monthly to periodic. The proposed change does not result in any changes in hardware or methods of operation. The details pertaining to the review of overtime are not considered in the safety analysis and cannot initiate or affect the mitigation of an accident in any way. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The change will not impose any new or different requirements or eliminate any existing requirements. 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 change involve a significant reduction in a margin of safety?

The proposed change does not impact the margin of safety since the margin of safety is not dependent on who performs the overtime review, nor on the frequency at which the review is performed. The safety analysis assumptions will still be maintained. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Less Restrictive Change L.7

The details specifying the actions to be taken in the event a Safety Limit is violated are deleted from the Specifications. The details regarding notification and reporting to the Commission are unnecessary, since reporting requirements are delineated in 10 CFR 50.72 and 50.73. The details regarding onsite notification requirements and review of the report by PORC [Plant Operations Review Committee] and NSRB [Nuclear Safety Review Board] are unnecessary, since plant policies and procedures already provide guidance on onsite notification and review of reports by these committees. Furthermore, these notification and reporting requirements are beyond the criteria of 10 CFR 50.36(c)(5) for inclusion in the Administrative Controls Section of the TS, and programmatic controls regarding actions to be taken for Safety Limit violations are not included in NUREG–1431. The proposed changes are provided in Table 1 as Item 30a.

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

The proposed changes delete the details regarding actions to be taken in the event of a Safety Limit violation. The proposed change does not result in any changes in hardware or methods of operation. The details pertaining to notification and reporting of Safety Limit violations are not considered in the safety analysis and cannot initiate or affect the mitigation of an accident in any way. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The change will not impose any new or different requirements or eliminate any existing requirements. 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 change involve a significant reduction in a margin of safety?

The proposed change does not impact the margin of safety since the margin of safety is not dependent on notification and reporting of Safety Limit violations. The safety analysis assumptions will still be maintained. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Relocation of Requirements

The proposed changes designated as ''Relocated'' (R) technical changes involve the relocation of existing TS requirements or details to other licensee-controlled documents such as the UFSAR [Updated Final Safety Analysis Report], TRM [Technical Requirements Manual], ODCM [Offsite Dose Calculation Manual], or OQAP [Operational Quality Assurance Plan]. Future modification of relocated Administrative Controls requirements is adequately controlled by regulatory requirements such as 10 CFR 50.59 and 10 CFR 50.54. The proposed changes include Items 4, 12, 13, 15, 22, 25, 29, 31, 32, 40, 41, 42, 44, 46, 49, 52, 55, 58, 59, 68, 75, 96, 112, 117, 118, and 126 in Table 1.

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

The proposed changes relocate certain details from the TS to the UFSAR, TRM, OQAP, or other licensee-controlled documents. These licensee-controlled documents containing the relocated information will be maintained in accordance with 10 CFR 50.59 or 10 CFR 50.54, as appropriate. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e) and the plant procedures and other licensee-controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Since any changes to the UFSAR, TRM, OQAP, or other licensee-controlled documents will be evaluated per 10 CFR 50.59 or 10 CFR 50.54, such changes will not involve a significant increase in the probability or consequences of an accident previously evaluated. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not involve physical alteration of the plant (no new or different type of equipment will be installed) or change in the methods governing normal plant operation. The proposed changes will not impose or eliminate any requirements and adequate control of the information will be maintained. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. In addition, the details to be relocated from the TS to the UFSAR, TRM, OQAP, or other licensee-controlled documents are the same as in the existing TS. Since any future change to these details in the UFSAR, TRM, OQAP, or other licensee-controlled documents will be evaluated per the requirements of 10 CFR 50.59 or 10 CFR 50.54, as appropriate, such changes would not involve a significant reduction in a margin of safety. Based on 10 CFR 50.92, the existing requirement for NRC review and approval of revisions to these details proposed for relocation does not have a specific margin of safety upon which to evaluate. However, since the proposed changes are consistent with NUREG–1431, which was approved by the NRC Staff, revising the TS to reflect the approved level of detail ensures no significant reduction in the margin of safety.

The NRC staff has reviewed the licenses's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Attorney for licensee: A. H. Gutterman, Esq., Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

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: December 10, 2001.

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) 4.0.1 and 4.0.3 from the current South Texas Project (STP) TS format to the Improved TS format. In addition, the licensee has proposed that a Bases Control Program be incorporated into Section 6.0 of the TSs in order to (1) specify an administrative process for making changes to the TS bases, (2) delineate what kinds of changes can be made to the TS Bases without prior NRC approval, and (3) to provide for consistency between the TS Bases and the STP Final Safety Analysis Report. TS 4.0.3 would also be changed to reflect Technical Specification Task Force (TSTF) 358, Revision 6, changes 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 surveillance interval, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater.'' The following requirement would be added to TS 4.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

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

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

The proposed change involves rewording of the existing Technical Specifications [4.0.1 and 4.0.3] to be consistent with NUREG–1431, Revision 2. These modifications involve no technical changes to the existing Technical Specifications. As such, these changes are administrative in nature and do not affect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change involves incorporation of the NUREG–1431, Revision 2, Bases Control Program requirements into the STP Technical Specifications. These modifications involve no technical changes to the existing Technical Specifications. As such, these changes are administrative in nature and do not affect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change involves rewording of the existing Technical Specifications [4.0.1 and 4.0.3] to be consistent with NUREG–1431, Revision 2. The change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change involves incorporation of the NUREG–1431, Revision 2, Bases Control Program requirements into the STP Technical Specifications. The changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change involves rewording of the existing Technical Specifications [4.0.1 and 4.0.3] to be consistent with NUREG–1431, Revision 2. The changes are administrative in nature and will not involve any technical changes. The changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. Also, since these changes are administrative in nature, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

The proposed change involves incorporation of the NUREG–1431, Revision 2, Bases Control Program requirements into the STP Technical Specifications. The changes are administrative in nature and will not involve any technical changes. The changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. Also, since these changes are administrative in nature, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

With regard to the changes associated with TSTF–358, Revision 6, 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 December 10, 2001.

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

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

The 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.

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

The proposed change does not involve a physical alteration of the plant (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.

3. The proposed change does not involve a significant reduction in 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 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: A. H. Gutterman, Esq., Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Robert A. Gramm.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: November 8, 2001 (TS 01–06).

Brief description of amendments: The proposed amendment would revise a License Condition and the Technical Specifications (TS) for Sequoyah Units 1 and 2. The proposed change would delete License Condition 2.H, ''Reporting to the Commission,'' Administrative Control Section 6.6, ''Reportable Event Action,'' and Administrative Control Section 6.7, ''Safety Limit Violation.'' Because Administrative Control Section 6.6 is referenced in several Limiting Conditions for Operation (LCOs) and associated TS Bases, these LCOs and TS Bases would also be modified to remove those references.

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

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

These revisions govern the reporting of either site characteristics and past events or of events covered under current NRC regulations and the proposed amendment is administrative in nature. Therefore, it does not increase the probability or consequences of any accident previously evaluated because it does not affect the state of the plant in any physical manner.

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

The proposed amendment is strictly administrative and does not affect plant equipment or operational procedures. Therefore, it will not create any new or different accidents.

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

The proposed amendment affects the reporting to the Commission. As such, it does not affect personnel, public, or plant safety. Since the amendment will not affect the plant in a physical manner nor will it affect personnel, public, or plant safety, it will therefore not reduce the margin of safety.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant (SQN), Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: January 15, 2002 (TS 01–13).

Brief description of amendments: The proposed amendment would revise Technical Specification (TS) Section 4.0.5.c to provide an exception to the recommendations of Regulatory Position c.4.b of NRC Regulatory Guide 1.14, Revision 1, ''Reactor Coolant Pump Flywheel Integrity,'' dated August 1975. This change is in accordance with Improved Standard TS Generic Change Traveler TSTF–237, Revision 1, Westinghouse Electrical Corporation Topical Report WCAP–14535A, ''Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination.''

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

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

An integral part of the Reactor Coolant System (RCS) in a pressurized water reactor is the RCP [reactor coolant pump]. The RCP ensures an adequate cooling flow rate by circulating large volumes of the primary coolant water at high temperature and pressure through the RCS. Following an assumed loss of power to the RCP motor, the flywheel, in conjunction with the impeller and motor assembly, provides sufficient rotational inertia to assure adequate core cooling flow during RCP coastdown.

Westinghouse Electric Corporation Topical Report WCAP–14535A, ''Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination,'' dated November 1996, provides the technical basis for the elimination of inspection requirements for RCP flywheels for all domestic Westinghouse plants. In the Safety Evaluation for WCAP–14535, dated September 1996, the NRC stated that the evaluation methodology described in WCAP–14535 is appropriate and the criteria are in accordance with the design criteria of RG 1.14.

RCP flywheel inspections have been performed for 20 years with no indications of service induced flaws. Flywheel integrity evaluations show a very high flaw tolerance for the RCP flywheels. Crack extension over a 60-year service life is negligible. Structural reliability studies have shown that eliminating inspections after 10 years of plant life will not significantly change the probability of failure.

The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed change does not alter or prevent the ability of structures, systems, and components (SSC) from performing their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the SQN Updated Final Safety Analysis Report (UFSAR). The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated in the SQN UFSAR.

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

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

The proposed change does not modify the design or function of the RCP flywheels. Based upon the results of WCAP–14535A, no new failure mechanisms will be introduced by the revised RCP Flywheel Inservice Inspection Program. As presented in WCAP–14535A, detailed stress analysis and risk assessments have been performed that indicate that there would be no change in the probability of failure for RCP flywheels if all inspections were eliminated. In addition, the flywheel integrity evaluations show that RCP flywheels exhibit a very high tolerance for the presence of flaws.

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

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

There is no significant mechanism for in-service degradation of the flywheels since they are isolated from the primary coolant environment. Additionally, WCAP–14535A analyses have shown there is no significant deformation of the flywheels even at maximum overspeed conditions. Likewise, the results of RCP flywheel inspections performed throughout the industry and at SQN identified no indications that would affect flywheel integrity.

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

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: December 26, 2001.

Brief description of amendments: The proposed amendments would revise Technical Specifications (TS) 5.5.16, ''Containment Leakage Rate Testing Program'' to allow for a one-time extension of the current interval between the Type A tests from 10 to 15 years.

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

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

Response: No.

The proposed revision to Technical Specifications adds a one time extension to the current interval for Type A testing (10CFR50, Appendix J, Option B, Integrated Leak Rate Testing). The current test interval of 10 years, based on past performance, would be extended on a one time basis to 15 years from the last Type A test. The proposed extension to Type A testing does not involve a significant increase in the consequences of an accident since research documented in NUREG–1493, ''Performance-Based Containment System Leakage Testing Requirements,'' September 1995, has found that, generically, very few potential containment leakage paths are not identified by Type B and C tests. The NUREG concluded that reducing the Type A testing frequency to one per twenty years was found to lead to an imperceptible increase in risk. A high degree of assurance is provided through testing and inspection that the containment will not degrade in a manner detectable only by Type A testing. The last Type A test show[s] leakage to be below acceptance criteria, indicating a very leak tight containment. Inspections required by the American Society of Mechanical Engineers (ASME) Code [Boiler and Pressure Vessel Code] Section XI (Subsections IWE and IWL) and maintenance rule monitoring (10CFR50.65, ''Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants'') are performed in order to identify indications of containment degradation that could affect that leak tightness. Type B and C testing required by Technical Specifications will identify any containment opening such as valves that would otherwise be detected by the Type A tests. These factors show that a Type A test extension will not represent a significant increase in the consequences of an accident.

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

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

Response: No.

The proposed revision to Technical Specifications adds a one time extension to the current interval for Type A testing (10CFR50, Appendix J, Option B, Integrated Leak Rate Testing). The current test interval of 10 years, based on past performance, would be extended on a one time basis to 15 years from the last Type A test. The proposed extension to Type A testing cannot create the possibility of a new or different type of accident since there are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure mode creating an accident or affecting the mitigation of an accident.

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

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

Response: No.

The proposed revision to Technical Specifications adds a one time extension to the current interval for Type A testing (10CFR50, Appendix J, Option B, Integrated Leak Rate Testing). The current test interval of 10 years, based on past performance, would be extended on a one time basis to 15 years from the last Type A test. The proposed extension to Type A testing will not significantly reduce the margin of safety. The NUREG–1493, ''Performance-Based Containment System Leakage Testing Requirements,'' September 1995, generic study of the effects of extending containment leakage testing found that a 20 year extension in Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG–1493 found that, generically, the design containment leakage rate contributes about 0.1 percent to the individual risk and that the decrease in Type A testing frequency would have a minimal affect on this risk since 95% of the potential leakage paths are detected by Type C testing. Regular inspections required by the American Society of Mechanical Engineers (ASME) Code Section XI (Subsections IWE and IWL) and maintenance rule monitoring (10CFR50.65, ''Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants'') will further reduce the risk of a containment leakage path going undetected.

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

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

Attorney for licensee: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

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

Date of application request: December 6, 2001.

Description of amendment request: The proposed amendment would revise Required Actions for Limiting Conditions for Operation (LCOs) 3.3.1, ''Reactor Trip (RTS) Instrumentation;'' 3.3.9, ''Boron Dilution Mitigation System (BDMS);'' 3.4.5, ''RCS Loops—MODE 3;'' 3.4.6, ''RCS Loops—MODE 4;'' 3.4.7, ''RCS Loops—MODE 5, Loops Filled;'' 3.4.8, ''RCS Loops—MODE 5, Loops Not Filled;'' 3.8.2, ''AC Sources—Shutdown;'' 3.8.5, ''DC Sources—Shutdown;'' 3.8.8, ''Inverters—Shutdown;'' 3.8.10, ''Distribution Systems—Shutdown;'' 3.9.3, ''Nuclear Instrumentation;'' 3.9.5, ''Residual Heat Removal (RHR) and Coolant Circulation—High Water Level;'' and 3.9.6, ''Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level'' in the Callaway Plant Technical Specifications (TSs). The Required Actions proposed to be revised require suspension of operations involving positive reactivity additions or reactor coolant system (RCS) boron concentration reductions. In addition, the proposed amendment would revise Notes, for several of the LCOs, that preclude reductions in RCS boron concentration. This amendment would revise these Required Actions and LCO Notes to allow small, controlled, safe insertions of positive reactivity, but limits the introduction of positive reactivity such that compliance with the required shutdown margin or refueling boron concentration limits will still be satisfied.

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.

Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes. The RTS instrumentation and reactivity control systems will be unaffected. Protection systems will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained.

The probability and consequences of accidents previously evaluated in the FSAR [Final Safety Analysis Report] are not adversely affected because the changes to the Required Actions and LCO Notes assure the limits on SDM [shutdown margin] and refueling boron concentration continue to be met, consistent with the analysis assumptions and initial conditions included within the safety analysis and licensing basis. The activities covered by this amendment application are routine operating evolutions. The proposed changes do not reduce the capability of reborating the RCS.

The proposed changes will not involve a significant increase in the probability of any event initiators. The initiating event for an inadvertent boron dilution event, as discussed in FSAR Section 15.4.6, is a failure in the reactor makeup control system (RMCS) or operator error such that inventory makeup with the incorrect boron concentration enters the RCS by way of the CVCS [chemical volume and control system] mixing tee. Since the RMCS design is unchanged, there will be no initiating event frequency increase associated with equipment failures. However, there could be an increased exposure time per operating cycle to potential operator errors during TS Conditions that, heretofore, prohibited positive reactivity additions. As such, the RTS Instrumentation, BDMS, and RCS Loops TS Bases changes from TSTF–286, Revision 2, have been augmented to preclude the introduction of reactor makeup water into the RCS via the CVCS mixing tee when one source range neutron flux channel (and, thus, the associated BDMS train) is inoperable or when no RCS loop is in operation. The equipment and processes used to implement RCS boration or dilution evolutions are unchanged and the equipment and processes are commonly used throughout the applicable MODES under consideration. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance. Required Action A.1 of LCO 3.3.9 limits the exposure to one inoperable BDMS train, which may be caused by an inoperable source range neutron flux channel. During the time the plant is in a TS Condition with a finite equipment restoration time, a single failure of the opposite train is not postulated. However, administrative controls have been added to this Action's Bases to highlight the need for operator awareness during all reactivity manipulations and to preclude introduction of reactor makeup water into the RCS.

The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR.

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

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

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. This amendment will not affect the normal method of plant operation or change any operating limits. The proposed changes merely permit the conduct of normal operating evolutions when additional controls over core reactivity are imposed by the Technical Specifications. The proposed changes do not introduce any new equipment into the plant or alter the manner in which existing equipment will be operated. The changes to operating procedures are minor, with clarifications provided that required limits must continue to be met. No performance requirements or response time limits will be affected. These changes are consistent with assumptions made in the safety analysis and licensing basis regarding limits on SDM and refueling boron concentration.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this amendment. There will be no adverse effect or challenges imposed on any safety-related system as a result of this amendment.

This amendment does not alter the design or performance of the 7300 Process Protection System, Nuclear Instrumentation System, or Solid State Protection System used in the plant protection systems.

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

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

The proposed changes do not alter the limits on SDM or refueling boron concentration. The nominal trip setpoints specified in the Technical Specifications Bases and the safety analysis limits assumed in the transient and accident analyses are unchanged. None of the acceptance criteria for any accident analysis is changed.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor ($F\Delta H$), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

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

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

Attorney for licensee: John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Stephen Dembek.

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

Date of application request: December 13, 2001.

Description of amendment request: The amendment would revise the Limiting Condition for Operation (LCO) 3.5.5, Required Action A.1 for the LCO, and Surveillance Requirement 3.5.5.1 in Technical Specification (TS) 3.5.5, ''Seal Injection Flow.'' The revision would replace the flow and differential pressure limits for the reactor coolant pump (RCP) seal injection flow stated in TS 3.5.5 by limits in Figure 3.5.5–1 that would be added to TS 3.5.5.

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

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

Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes. The RTS [reactor trip system] instrumentation and reactivity control systems will be unaffected. Protection systems will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained.

The probability and consequences of accidents previously evaluated in the FSAR [Final Safety Analysis Report] are not adversely affected because the changes continue to assure the analysis assumptions and initial conditions included within the safety analysis and licensing basis are satisfied.

The proposed changes will not involve a significant increase in the probability of any event initiators. The initiating event for a loss of coolant accident, as discussed in FSAR Section 15.6.5, is a break in the RCS [reactor coolant system] piping. Since the RCS piping design is unchanged, there will be no initiating event frequency increase associated with pipe breaks. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance.

The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR.

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

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

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. This amendment will not affect the normal method of plant operation. The proposed changes do not introduce any new equipment into the plant or alter the manner in which existing equipment will be operated. The changes to operating procedures are minor, with clarifications provided that required limits must continue to be met. No performance requirements or response time limits will be affected. These changes are consistent with assumptions made in the safety analysis and licensing basis regarding limits on RCP seal injection flow.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this amendment. There will be no adverse effect or challenges imposed on any safety-related system as a result of this amendment.

This amendment does not alter the design or performance of the 7300 Process Protection System, Nuclear Instrumentation System, or Solid State Protection System used in the plant protection systems. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes do not alter the input parameters listed in FSAR Table 15.6–9 and used in large break and small break LOCA [loss-of-coolant accident] peak cladding temperature analyses. The containment pressure and temperature analyses are not adversely impacted. The nominal reactor and ESFAS [engineered safety feature actuation system] trip setpoints (Technical Specification Bases Tables B 3.3.1–1 and B 3.3.2–1), reactor and ESFAS allowable values (Technical Specification Tables 3.3.1–1 and 3.3.2–1), and the safety analysis limits assumed in the transient and accident analyses (FSAR Table 15.0–4) are unchanged. None of the acceptance criteria for any accident analysis is changed.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protective functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor (FΔH), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

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

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

Attorney for licensee: John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: April 3, 2001 as supplemented by letters dated October 22 and December 18, 2001. The April 3, 2001, amendment application was previously noticed in the **Federal Register** on May 2, 2001 (66 FR 22036).

Description of amendment request: The supplemental letter of October 22, 2001, added the following change to the technical specifications (TSs): revise TS Section 5.6.5 by adding TS 2.1.1 on reactor core safety limits on the existing list of core operating limits for each reload cycle that are documented in the Core Operating Limits Report (COLR). This proposed change is being added to the previous changes requested by the licensee's letter of April 3, 2001. The amendment would make the following changes to the TSs:

(1) Revise Safety Limit 2.1.1 by replacing Figure 2.1.1–1, ''Reactor Core Safety Limits,'' with a reference to limits being specified in the Core Operating Limits Report (COLR) and by adding two reactor core safety limits on departure from nucleate boiling ratio (DNBR) and peak fuel centerline temperature.

(2) Revise Note 1 on the over temperature ΔT in Table 3.3.1–1 of TS 3.3.1, ''Reactor Trip System Instrumentation,'' by replacing values of parameters with a reference to the values being specified in the COLR and correcting the expression for one term in the inequality for over temperature ΔT.

(3) Revise Note 2 on the overpower ΔT in Table 3.3.1–1 by replacing values of parameters with a reference to the values being specified in the COLR.

(4) Replace the limits for the reactor coolant system (RCS) pressure and average temperature with a reference to the limits being specified in the COLR for Limiting Condition for Operation (LCO) 3.4.1 and Surveillance Requirements (SRs) 3.4.1.1 and 3.4.1.2.

(5) Add the phrase ''and greater than or equal to the limit specified in the COLR'' to the RCS total flow rate in LCO 3.4.1 and SRs 3.4.1.3 and 3.4.1.4.

(6) Move items a. and b. to the left in the Note to the applicability in LCO 3.4.1.

(7) Revise TS Section 5.6.5 by adding TS 2.1.1 on reactor core safety limits, TS 3.3.1 on over temperature and overpower ΔT trip setpoints, and TS 3.4.1 on RCS pressure, temperature, and flow limits to the existing list of core operating limits for each reload cycle that are documented in the COLR and revising the list of topical reports in the COLR that represent the analytical methods approved by the Commission to determine core operating limits.

The proposed changes remove cycle-specific parameter limits and relocate them to the COLR, but they do not change any of the limits. The changes add more specific requirements regarding DNBR limit and peak fuel centerline temperature limit to the TSs, revise the list of topical reports in the list of NRC-approved analytical methods, correct one term of an expression, and move terms in a Note to the mode applicability for an LCO.

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 are programmatic and administrative in nature which do not physically alter safety related systems, nor affect the way in which safety related systems perform their functions. More specific requirements regarding the safety limits (i.e., DNBR limit and peak fuel centerline temperature limit) are being imposed in TS 2.1.1, ''Reactor Core Safety Limits,'' which replace the Reactor Core Safety Limits figure and are consistent with the values stated in the USAR [Updated Safety Analysis Report]. The proposed changes remove the cycle-specific parameter limits from TS 3.4.1 and relocate them to the COLR which do not change plant design or affect system operating parameters. In addition, the minimum limit for RCS total flow rate is being retained in TS 3.4.1 to assure that a lower flow rate than reviewed by the NRC will not be used. The proposed changes do not, by themselves, alter any of the parameter limits. The removal of the cycle-specific parameter limits from the TS does not eliminate existing requirements to comply with the parameter limits. The existing TS Section 5.6.5b, COLR Reporting Requirements, continues to ensure that the analytical methods used to determine the core operating limits meet NRC reviewed and approved methodologies. The existing TS Section 5.6.5c, COLR Reporting

Requirements, continues to ensure that applicable limits of the safety analyses are met.

The proposed changes to reference only the Topical Report number and title do not alter the use of the analytical methods used to determine core operating limits that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the COLR without having to submit an amendment to [the TS of] the operating license. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval.

Although the relocation of the cycle-specific parameter limits to the COLR would allow revision of the affected parameter limits without prior NRC approval, there is no significant effect on the probability or consequences of an accident previously evaluated. Future changes to the COLR parameter limits could result in event consequences which are either slightly less or slightly more severe than the consequences for the same event using the present parameter limits. The differences would not be significant and would be bounded by the existing requirement of TS Section 5.6.5c to meet the applicable limits of the safety analyses.

The cycle-specific parameter limits being transferred from the TS to the COLR will continue to be controlled under existing programs and procedures. The USAR accident analyses will continue to be examined with respect to changes in the cycle-dependent parameters obtained using NRC reviewed and approved reload design methodologies, ensuring that the transient evaluation of new reload designs are bounded by previously accepted analyses. This examination will continue to be performed pursuant to 10 CFR 50.59 requirements ensuring that future reload designs will not involve a significant increase in the probability or consequences of an accident previously evaluated. Additionally, the proposed changes do not allow for an increase in plant power levels, do not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes do not change the types or increase the amounts of any effluents released offsite.

[The proposed changes to the expression of the $f_1(\Delta I)$ term, which is in the over temperature ΔT inequality, clarifies and corrects the term. Moving the terms in a Note to the LCO mode applicability is an administrative action.]

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

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

[The proposed changes are programmatic and administrative in nature which do not physically alter safety related systems, nor affect the way in which safety related systems perform their functions.]

The proposed changes that retain the minimum limit for RCS total flow rate in the TS, and that relocate certain cycle-specific parameter limits from the TS to the COLR, thus removing the requirement for prior NRC approval of revisions to those parameters, do not involve a physical change to the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There are no changes being made to the parameters within which the plant is operated, other than their relocation to the COLR. There are no setpoints affected by the proposed changes at which protective or mitigative actions are initiated. The proposed changes will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alteration in the procedures which ensure the plant remains within analytical limits is being proposed, and no change is being made to the procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced.

The proposed changes to reference only the Topical Report number and title do not alter the use of the analytical methods used to determine core operating limits that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the COLR without having to submit an amendment to [the TS of] the operating license. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval.

Relocation of cycle-specific parameter limits has no influence or impact on, nor does it contribute in any way to the possibility of a new or different kind of accident. The relocated cycle-specific parameter limits will continue to be calculated using the NRC reviewed and approved methodology. The proposed changes do not alter assumptions made in the safety analysis and operation within the core operating limits will continue.

[The proposed changes to the expression of the $f_1(\Delta I)$ term, which is in the over temperature ΔT inequality, clarifies and corrects the term.]

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

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

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed changes [are programmatic and administrative in nature and] do not physically alter safety related systems, nor does it [a]ffect the way in which safety-related systems perform their functions. The setpoints at which protective actions are initiated are not altered by the proposed changes.

Therefore, sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. As the proposed changes to relocate cycle-specific parameter limits to the COLR will not affect plant design or system operating parameters, there is no detrimental impact on any equipment design parameter, and the plant will continue to operate within prescribed limits.

The development of cycle-specific parameter limits for future reload designs will continue to conform to NRC reviewed and approved methodologies, and will be performed pursuant to 10 CFR 50.59 to assure that plant operation [is] within cycle-specific parameter limits.

The proposed changes to reference only the Topical Report number and title do not alter the use of the analytical methods used to determine core operating limits that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of [the] current Topical Reports to support limits in the COLR without having to submit an amendment to [the TS of] the operating license. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval.

[The proposed changes to the expression of the $f_1(\Delta I)$ term, which is in the over temperature ΔT inequality, clarifies and corrects the term. Moving the terms in a Note to the LCO mode applicability is an administrative action.]

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

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

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

NRC Section Chief: Stephen Dembek.

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.*

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

Date of application for amendment: July 9, 2001.

Brief description of amendment: The amendment revised the Administrative Controls Section of the Technical Specifications to provide consistency with the changes to Title 10 of the Code of Federal Regulations (10 CFR), Section 50.59, which were published in the **Federal Register** on October 4, 1999 (64 FR 53582). Specifically, the amendment replaced the term ''safety evaluation'' with ''10 CFR 50.59 evaluation'' and the term ''unreviewed safety question'' with ''requires NRC [Nuclear Regulatory Commission] approval pursuant to 10 CFR 50.59.''

Date of issuance: January 22, 2002.

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

Amendment No.: 239.

Facility Operating License No. DPR–50. Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 22, 2001 (66 FR 44162).

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

No significant hazards consideration comments received: No.

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of application for amendment: October 31, 2001.

Brief description of amendment: The amendment deletes Technical Specification 5.5.3, ''Post-Accident Sampling,'' eliminating the requirement to have and maintain the Post-Accident Sampling System at H. B. Robinson. The amendment also deletes Condition 3.G.(4) of the Operating License.

Date of issuance: January 14, 2002.

Effective date: January 14, 2002.

Amendment No. 192.

Facility Operating License No. DPR–23. Amendment revises the License and Technical Specifications.

*Date of initial notice in **Federal Register:*** December 12, 2001 (66 FR 64286) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 14, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 30, 2001.

Brief description of amendment: The amendment deletes requirements from the Technical Specifications (and, as applicable, other elements of the licensing bases) to maintain a Post-Accident Sampling System.

Date of issuance: January 14, 2002.

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

Amendment No.: 108.

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

*Date of initial notice in **Federal Register:*** December 12, 2001 (66 FR 64287).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 11, 2001, as supplemented on September 26 and November 16, 2001.

Brief description of amendment: The amendment approves a change to Technical Specifications 1.12, ''Core Alteration;'' 3.9.1, ''Refueling Operations—Boron Concentration;'' 3.9.2, ''Refueling Operations—Instrumentation;'' and 3.9.11, ''Refueling Operations—Water Level—Reactor Vessel.'' The amendment also revises the Technical Specifications Bases to reflect the changes to the definitions.

Date of issuance: January 11, 2002.

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

Amendment No.: 263.

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

*Date of initial notice in **Federal Register:*** June 12, 2001 (66 FR 31705).

The September 26 and November 16, 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand beyond the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 11, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 23, 2001, as supplemented December 20, 2001.

Brief description of amendment: The amendment revised Technical Specification Surveillance Requirement 3.8.4.1 to support replacement of the station batteries. The amendment will allow for separate required terminal voltage values for the new 31 and 32 station batteries.

Date of issuance: January 17, 2002.

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

Amendment No.: 209.

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

*Date of initial notice in **Federal Register:*** November 28, 2001 (66 FR 59503).

The December 20, 2001, 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 January 17, 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: October 12, 2001.

Brief description of amendments: The amendments delete TS 6.8.3 requiring a program for post accident sampling, and thereby eliminate the requirements to have and maintain Post Accident Sampling System at Donald C. Cook Nuclear Plant, Units 1 and 2.

Date of issuance: January 16, 2002.

Effective date: As of the date of issuance and shall be implemented within 120 days.

Amendment Nos.: 261—Unit 1, 244—Unit 2.

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 64295)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 16, 2002.

No significant hazards consideration comments received: No.

Indiana Michigan Power Company, Docket No. 50–315, Donald C. Cook Nuclear Plant, Unit 1, Berrien County, Michigan

Date of application for amendment: November 19, 2001.

Brief description of amendment: The amendment would revise the action statement for Technical Specification (TS) 3.3.3.5, ''Remote Shutdown Instrumentation,'' to add a statement that the provisions of TS 3.0.4 are not applicable.

Date of issuance: January 16, 2002.

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

Amendment No.: 262.

Facility Operating License No. DPR–58: Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 12, 2001 (66 FR 64295).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 16, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 22, 2001.

Brief description of amendment: The amendment revised the KNPP TS 6.14, ''Post Accident Sampling and Monitoring,'' and thereby eliminate the requirements to have and maintain the Post Accident Sampling System. Although TS 6.14's title contains the word ''monitoring,'' elimination of this TS does not eliminate the post-accident monitoring instrumentation from KNPP TS. These instruments are contained in KNPP TS section 3.5, which are listed in TS Table 3.5–6, ''Accident Monitoring Instrumentation Operating Conditions for Indication.''

Date of issuance: January 16, 2002.

Effective date: As of the date of issuance and shall be implemented within 120 days.

Amendment No.: 160.

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

*Date of initial notice in **Federal Register:*** December 12, 2001 (66 FR 64299).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 16, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 30, 2001.

Brief description of amendment: The amendment eliminates local suppression pool temperature limits from the Updated Safety Analysis Report as the basis for limiting suppression pool mechanical loads due to unstable steam condensation during safety relief valve actuations.

Date of issuance: January 18, 2002.

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

Amendment No.: 126.

Facility Operating License No. DPR–22. Amendment revised the licensing basis.

*Date of initial notice in **Federal Register:*** June 27, 2001 (66 FR 34286).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 18, 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: May 15, 2001, as supplemented by letters dated June 14 and November 21, 2001.

Brief description of amendment: The amendment: (1) replaced the titles of Manager—Fort Calhoun Station and Vice President with generic titles, (2) relocated the requirements for the Plant Review Committee (PRC) and the Safety Audit and Review Committee (SARC) to the Fort Calhoun Station Quality Assurance Program, (3) relocated the requirements for procedure controls and records retention to the Fort Calhoun Station Quality Assurance Program, (4) enhanced and clarified the qualification and training requirements for individuals who perform licensed operator functions, (5) incorporated the Westinghouse/CENP definition of azimuthal power tilt, and (6) eliminated specific mailing address and reporting requirements that are redundant to Title 10 of the Code of Federal Regulations.

Date of issuance: January 11, 2002.

Effective date: January 11, 2002, and shall be implemented within 30 days from the date of issuance.

Amendment No.: 202.

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

*Date of initial notice in **Federal Register:*** June 27, 2001 (66 FR 34287). The June 14 and November 21, 2001, supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 11, 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: April 2, 2001.

Brief description of amendment: This amendment revises the Technical Specifications (TSs) to relocate TS Sections 3/4.9.4, ''Refueling Operations, Decay Time;'' 3/4.9.5, ''Refueling Operations, Communications;'' 3/4.9.6, ''Refueling Operations, Refueling Platform;'' and 3/4.9.7, ''Refueling Operations, Crane Travel—Spent Fuel Storage Pool'' and the associated TS Bases pages to the Hope Creek Generating Station Updated Final Safety Analysis Report.

Date of issuance: January 17, 2002.

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

Amendment No.: 137.

Facility Operating License No. NPF–57: This amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 16, 2001 (66 FR 27177).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 17, 2002.

No significant hazards consideration comments received: No.

Rochester Gas and Electric Corporation, Docket No. 50–244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of application for amendment: October 25, 2001.

Brief description of amendment: The amendment deletes Technical Specification Section 5.5.3, ''Post Accident Sampling Program'', and thereby eliminates the requirements to have and maintain the Post-Accident Sampling System.

Date of issuance: January 17, 2002.

Effective date: January 17, 2002.

Amendment No.: 81.

Facility Operating License No. DPR–18: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 12, 2001 (66 FR 64300).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 17, 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: August 24, 2001, as supplemented by e-mail dated November 16, 2001.

Brief description of amendments: The amendments decrease the calculated peak containment internal pressure for the design basis loss-of-coolant accident and main steamline break from 55.1 to 45.9 psig and 56.6 to 56.5 psig, respectively, in Section 5.5.2.15, ''Containment Leakage Rate Testing Program,'' of the Technical Specifications.

Date of issuance: January 24, 2002.

Effective date: January 24, 2002, to be implemented within 60 days of issuance.

Amendment Nos.: Unit 2—182; Unit 3—173.

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

*Date of initial notice in **Federal Register:*** October 3. 2001 (66 FR 50472).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 24, 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: May 24, 2001.

Brief description of amendment: This amendment revises Technical Specifications Sections 4.2.2.2.e and g, and 4.2.2.4.e and g to adopt a modified methodology that relocates the heat flux hot channel factor, $F_Q(z)$, penalty for increasing $F_Q(z)$ versus burnup to a table in the Core Operating Limits Report. The amendment also increases the surveillance region of $F_Q(z)$ to be consistent with the current core design and provide assurance that the peak $F_Q(z)$ is monitored and evaluated near end of core life.

Date of issuance: January 24, 2002.

Effective date: January 24, 2002.

Amendment No.: 153.

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

*Date of initial notice in **Federal Register:*** July 25, 2001 (66 FR 38766).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 24, 2002.

No significant hazards consideration comments received: No.

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

Date of amendment request: May 9, 2001.

Brief description of amendments: The amendments consist of changes to the Technical Specifications, extending the emergency core cooling system accumulator's allowable outage time from 12 hours to 24 hours.

Date of issuance: January 10, 2002.

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

Amendment Nos.: Unit 1—135; Unit 2—124.

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

*Date of initial notice in **Federal Register:*** August 22, 2001 (66 FR 44176).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 10, 2002.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: October 31, 2001.

Brief description of amendments: The amendments delete the program requirements of Technical Specification 6.8.4.e, ''Post Accident Sampling.''

Date of issuance: January 14, 2002.

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

Amendment Nos.: 272 and 261.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the TSs.

Date of initial notice in ***Federal Register:*** December 12, 2001 (66 FR 64302).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 14, 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: October 31, 2001.

Brief description of amendment: The amendment deletes the program requirements of Technical Specification 5.7.2.6, ''Post Accident Sampling System.''

Date of issuance: January 14, 2002.

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

Amendment No.: 34.

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

*Date of initial notice in **Federal Register:*** December 12, 2001 (66 FR 64304).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 14, 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: May 14, 2001.

Brief description of amendment: The amendment incorporates part of TSTF–51, Revision 2 into the Watts Bar Technical Specifications (TS). TSTF–51 allows revising the TS to eliminate engineered safety features operability requirements that do not involve the movement of irradiated fuel during core alterations.

Date of issuance: January 22, 2002.

Effective date: As of the date of issuance and shall be implemented prior to entering Mode 6 for the Cycle 4 refueling outage.

Amendment No.: 35.

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

Date of initial notice in ***Federal Register:*** July 25, 2001 (66 FR 38768).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 22, 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: May 14, 2001.

Brief description of amendment: The amendment revises Technical Specification section 3.3.5 ''Loss of Power (LOP) Diesel Generator Start Instrumentation,'' to increase the time delay setting of the 6.9kV shutdown board degraded voltage relays from a nominal 6 seconds to 10 seconds.

Date of issuance: January 23, 2002.

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

Amendment No.: 36.

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

Date of initial notice in ***Federal Register:*** July 25, 2001 (66 FR 38767).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 23, 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: October 2, 2001.

Brief description of amendments: The amendments delete Technical Specification (TS) 5.5.3, ''Post Accident Sampling System,'' and thereby eliminate the requirements to have and maintain the Post Accident Sampling System (PASS) at Comenche Peak Steam Electric Station. In addition, the amendments revise TS 5.5.2, ''Primary Coolant Sources Outside Containment,'' to reflect the elimination of PASS.

Date of issuance: January 15, 2002.

Effective date: As of the date of issuance and shall be implemented by March 15, 2003.

Amendment Nos.: 91 and 91.

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

Date of initial notice in ***Federal Register:*** December 12, 2001.

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

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 29th day of January, 2002.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 02–2567 Filed 2–4–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[NUREG–1569]

Solicitation of Comments on a Draft Standard Review Plan (NUREG–1569) for Staff Reviews for in Situ Leach Uranium Extraction License Applications

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability; Opportunity for comment.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is soliciting comments from interested parties on a Draft Standard Review Plan (NUREG–1569) which provides guidance for staff safety and environmental reviews of applications to develop and operate uranium in situ leach facilities. An NRC Materials License is required, under the provisions of Title 10 of the Code of Federal Regulations, part 40 (10 CFR part 40), Domestic Licensing of Source Material, in conjunction with uranium extraction by in situ leach extraction techniques.

The applicant for a license is required to provided detailed information on the facilities, equipment, and procedures used and an Environmental Report that discusses the effects of proposed operations on the health and safety of the public and on the environment. This information, and the licensee's Environmental Report, are used by the NRC staff to determine whether the proposed activities will be protective of public health and safety and the environment.

This draft Standard Review Plan provides the NRC staff with specific guidance on performing reviews of this information and will be used to ensure a consistent quality and uniformity of staff reviews. Each section in the review plan provides guidance on what is to be reviewed, the basis for the review, how the staff review is to be accomplished, what the staff will find acceptable in a demonstration of compliance with the regulations, and the conclusions that are sought regarding the applicable sections in 10 CFR part 40, Appendix A. The Standard Review Plan is also intended to improve the understanding of interested members of the public, and the uranium recovery industry, of the staff review process.

A draft of NUREG–1569 was issued in October 1997 for public comment. This draft of NUREG–1569 incorporates the staff responses to comments and the results of Commission policy decisions affecting uranium recovery issues, which are described in NRC Regulatory Issue Summary 2000–23, dated November 30, 2000.

Opportunity to Comment: Interested parties are invited to comment on the standard review plan. A final standard review plan will be prepared after the NRC staff has evaluated comments received on the draft standard review plan. Written comments must be received prior to April 22, 2002. Comments on the draft review plan should be sent to the Chief, Rules and Directives Branch, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

A copy of the Draft Standard Review Plan (NUREG–1569) may be obtained by writing to the Reproduction and Distribution Services Branch, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, or e-mail *distribution@nrc.gov.*

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