Week of December 2, 2002—Tentative

Wednesday, December 4, 2002

10 a.m.—Briefing on Decommissioning Bankruptcy Issues (Closed—Ex. 4 & 9)

Week of December 9, 2002—Tentative

There are no meetings scheduled for the Week of December 9, 2002.

Week of December 16, 2002—Tentative

Tuesday, December 17, 2002

9:30 a.m.—Briefing on policy options and recommendations for revising the NRC's process for handling discrimination issues (public meeting) (Contact: Ho Nieh, 301–415–1721)

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

Wednesday, December 18, 2002

9:30 a.m.—Meeting with advisory committee on nuclear waste (ACNW) (public meeting) (Contact: John Larkins, 301–415–7360)

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

Week of December 23, 2002—Tentative

There are no meetings scheduled for the Week of December 23, 2002.

Week of December 30, 2002—Tentative

There are no meetings scheduled for the Week of December 30, 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: R. Michelle Schroll (301) 415–1662.

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Additional Information: By a vote of 5–0 on November 20, 2002, the Commission determined pursuant to U.S.C. 552b(e) and 9.107(a) of the Commission's rules that ''Affirmation of (a) Final Rule on Decommissioning Trust Provisions, (b) Final Rule: Material Control and Accounting Amendments, (c) Dominion Nuclear Connecticut, Inc. (Millstone Nuclear Power Station, Unit No. 3; Facility Operating License NPF–49), and (d) Pacific Gas & Electric Co. (Diablo Canyon Power Plant Independent Spent Fuel Storage Installation); Petition to Suspend Proceeding Pending Comprehensive Review of Adequacy of Design and Operation Measures to Protect Against Terrorist Attack and Other Acts of Malice or Insanity'' be held on November 21, 2002, and on less than one week's notice to the public.

* * * * *

The NRC Commission Meeting Schedule can be found on the Internet at: *http://www.nrc.gov/what-we-do/policy-making/schedule.html.*

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This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 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: November 21, 2002.

R. Michelle Schroll,

Acting Technical Coordinator, Office of the Secretary.

[FR Doc. 02–30099 Filed 11–22–02; 12:06 pm]

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 November 1, 2002, through November 14, 2002. The last biweekly notice was published on November 12, 2002 (67 FR 68727).

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

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

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

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

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

By December 26, 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,[1] which is available at the Commission's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, 304–415–4737 or by e-mail to *pdr@nrc.gov.*

Arizona Public Service Company, et al., Docket No. STN 50–528, Palo Verde Nuclear Generating Station, Unit 1, Maricopa County, Arizona

Date of amendment request: September 26, 2002, as supplemented by letter dated October 23, 2002.

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

Description of amendment request: The amendment would revise Technical Specification (TS) 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' to clearly delineate the scope of the tube inspection required in the SG tubesheet region. TS 5.5.9 is in section 5, ''Administration Controls,'' of the TSs.

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

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

Response: No.

Arizona Public Service Company (APS) proposes to modify Palo Verde Nuclear Generating Station (PVNGS) Technical Specifications for Unit 1 to define the SG tube inspection scope. The PVNGS Unit 1 specific analysis takes into account the reinforcing effect the tubesheet has on the external surface of an expanded SG tube. Tube-bundle integrity will not be adversely affected by the implementation of the revised tube inspection scope. SG tube burst or collapse cannot occur within the confines of the tubesheet; therefore, the tube burst and collapse criteria of NRC Regulatory Guide (RG) 1.121 (Bases for Plugging Degraded PWR Steam Generator Tubes) are inherently met. Any degradation below the TEA (Tube Engagement Area) length is shown by analyses and test results to be acceptable, thereby precluding an event with consequences similar to a postulated tube rupture event.

Tube burst is precluded for cracks within the tubesheet by the constraint provided by the tubesheet. Thus, structural integrity is maintained by the tubesheet constraint. However, a 360-degree circumferential crack or many axially oriented cracks could permit severing of the tube and tube pullout from the tubesheet under the axial forces on the tube from primary to secondary pressure differentials. Testing was performed to define the length of non-degraded tubing that is sufficient to compensate for the axial forces on the tube and thus prevent pullout. This proposed amendment would encompass that length of non-degraded tubing for inspection.

In conclusion, incorporation of the revised inspection scope into PVNGS Unit 1 Technical Specifications maintains existing design limits and therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

Tube-bundle integrity is expected to be maintained during all plant conditions upon implementation of the proposed tube inspection scope. Use of this scope does not introduce a new mechanism that would result in a different kind of accident from those previously analyzed. Even with the limiting circumstances of a complete circumferential separation of a tube occurring below the TEA length, SG tube pullout is precluded and leakage is predicted to be maintained within the Updated Final Safety Analysis Report limits during all plant conditions.

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

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

Response: No.

Upon implementation of the revised inspection scope, operation with potential cracking below the Inspection Extent length in the explansion region of the SG tubing meets the margin of safety as defined by RG 1.121 and RG 1.83 (Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes) and the requirements of General Design Criteria 14, 15, 31, and 32 of 10 CFR (part) 50. Accordingly, the proposed change does not involve a significant reduction in a margin of safety.

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

The NRC staff has reviewed the licensee's analysis and, based on 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 amendment involves no significant hazards consideration.

The above amendment was previously noticed in the **Federal Register** on October 3, 2002 (67 FR 62079), as an exigent circumstances TS amendment, based on the preliminary determination that the TS amendment was needed on or about October 25, 2002, to allow Unit 1 to restart from its refueling outage. On further consideration, it has been determined that the proposed TS amendment does not have to be issued before the restart of Unit 1. This notice supersedes and replaces the exigent circumstances TS amendment notice of October 3, 2002.

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.

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

Date of amendment request: September 26, 2002.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.9.1, ''Refueling Equipment Interlocks,'' to allow fuel movement to continue if the refueling interlocks become inoperable, and add two new alternative Required Actions for the condition when the refueling equipment interlocks are inoperable. Specifically, the proposed amendment would add Required Actions 3.9.1.A.2.1 to immediately block control rod withdrawal and 3.9.1.A.2.2 to perform a verification that all of the control rods are fully inserted. The proposed changes are similar to the proposed generic change that was provided in Technical Specifications Task Force (TSTF) Traveler, TSTF–225, revision 1, ''Fuel Movement With Inoperable Refueling Equipment Interlocks,'' dated November 22, 2000, for the NRC staff's review.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, 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 amendment to the Technical Specifications does not result in the alteration of the design, material, or construction standards that were applicable prior to the change. The same Refueling Interlocks instrumentation is used, and the control rod removal error and fuel assembly insertion error assumptions in the Updated Final Safety Analysis Report (UFSAR) chapter 15 analysis remain unchanged. The proposed additional Required Actions provide an equivalent level of assurance that fuel will not be loaded into a core cell with a control rod withdrawn as does the current TS Required Action. The proposed change will not result in the modification of any system interface that would increase the likelihood of an accident since these events are independent of the proposed change. The proposed amendment will not change, degrade, or prevent actions, or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the UFSAR. Therefore, the proposed amendment does not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

This change in the TS requirements does not alter the performance of the Refueling Equipment Interlocks. The change does not involve a change in plant design or to the analyzed condition of the reactor core during

refueling. The proposed new Required Actions will ensure that control rods are not withdrawn and cannot be inappropriately withdrawn because a block to control rod withdrawal is in place. Implementation of the proposed amendment does not create the possibility of a new of different kind of accident from any accident previously evaluated.

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

Response: No.

As discussed in the Bases for the affected TS requirements, inadvertent criticality is prevented during the loading of fuel provided all control rods are fully inserted. The refueling interlocks function to support the refueling procedures by preventing control rod withdrawal during fuel movement, and the inadvertent loading of fuel when a control rod is withdrawn. The proposed change will allow the refueling interlocks to be inoperable and fuel movement to continue, only if a control rod withdrawal block is in effect and all control rods are verified to be fully inserted. These proposed Required Actions provide an equivalent level of protection as the refueling interlocks by preventing a configuration which could lead to an inadvertent criticality event. The refueling procedures will continue to be supported by the proposed Required Actions because control rods cannot be withdrawn and as a result, fuel cannot be inadvertently loaded when a control rod is withdrawn. Plant and system response to an initiating event will remain in compliance within the assumptions of the safety analyses, and therefore, the margin of safety is not affected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: L. Raghavan.

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

Date of amendment request: September 26, 2002.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Surveillance Requirement (SR) 3.7.3.6 associated with the verification of the control room emergency filtration (CREF) system duct work unfiltered in-leakage. Specifically, the proposed amendment would add a note to SR 3.7.3.6 to allow crediting the performance of an integrated tracer gas test of the control room envelope while in the recirculation mode to satisfy the requirements of the surveillance.

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.

This license amendment proposes an alternative test for performing the CREF system surveillance associated with measuring the Control Room Envelope (CRE) unfiltered inleakage. The CREF system provides a configuration for mitigating radiological consequences of accidents; however, it does not involve the initiation of any previously analyzed accident. Therefore, the proposed change cannot increase the probability of any previously evaluated accident.

The CREF system provides a radiologically controlled environment from which the plant can be safely operated following a radiological accident. Design basis accident analyses conclude that radiological consequences are within the regulatory acceptance criteria. The current Technical Specifications (TS) surveillance (SR 3.7.3.6) measures inleakage from four sections of CREF system duct work outside the CRE that are at negative pressure during accident conditions. The proposed Tracer Gas test provides a measurement of CRE inleakage from all potential sources including the four sections of duct work. The use of Tracer Gas testing in accordance with the methods described in American Society of Testing and Materials (ASTM) standard E741 has been accepted by both the NRC and the industry. Measuring the CRE inleakage using Tracer Gas testing has no effect on the CREF system function. The results of Tracer Gas testing will be assessed in accordance with regulatory guidance and industry guidance and compliance with 10 CFR [part] 50, Appendix A, General Design Criterion (GDC)–19 will be demonstrated. Therefore, the proposed change does not significantly increase the radiological consequences of any previously evaluated accident. Based on the above, the proposed change does not significantly increase the probability or consequences of any accident previously evaluated.

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

The proposed change does not alter the design function or operation of the system involved. The CREF system will still provide protection to control room occupants in case of a significant radioactive release. The revised TS surveillance requirements provide an alternative test method that has been widely accepted for the measurement of CRE unfiltered inleakage. The proposed change does not introduce any new modes of plant or CREF system operation and does not involve physical modifications to the plant. Therefore, the proposed change does not create the potential for 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 proposed change to the Fermi 2 TS surveillance requirements does not affect the radiological release from a design basis accident nor the postulated dose to the control room occupants as a result of the accident. The alternate surveillance test requirements provide an acceptable approach for the measurement of CRE inleakage. Safety margins and analytical conservatisms are included in the analyses to ensure that all postulated event scenarios are bounded. The proposed TS requirements continue to ensure that the radiological consequences at the control room are below the corresponding regulatory guidelines and that compliance with GDC–19 is not affected. Therefore, the proposed changes will not result in a significant reduction in the margin of safety.

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

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

NRC Section Chief: L. Raghavan.

Duke Energy Corporation, et al., Docket No. 50–414, Catawba Nuclear Station, Unit 2, York County, South Carolina

Date of amendment request: October 10, 2002.

Description of amendment request: The amendment would allow Duke Energy Corporation to continue using the reactor coolant system cold leg elbow tap flow coefficient that was approved by Nuclear Regulatory Commission on an interim basis for Cycle 12 at Catawba Nuclear Station, Unit 2. No changes in Technical Specifications are necessary for this Amendment.

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

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

1. Involve a significant increase in the probability or consequences of an accident previously evaluated, or
2. Create the possibility of a new or different kind of accident from any accident previously evaluated, or
3. Involve a significant reduction in a margin of safety.

First Standard

The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. No component modification, system realignment, or change in operating procedure will occur which could affect the probability of any accident or transient. The revised cold leg elbow tap flow coefficients will not change the probability of actuation of any Engineered Safeguards Feature or other device. The actual Unit 2 RCS [reactor coolant system] flow rate will not change. Therefore, the consequences of previously analyzed accidents will not change as a result of the revised flow coefficients.

Second Standard

The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. No component modification or system realignment will occur which could create the possibility of a new event not previously considered. No change to any methods of plant operation will be required. The elbow taps are already in place, and are presently being used to monitor flow for Reactor Protection System purposes. They will not initiate any new events.

Third Standard

The proposed amendment will not involve a significant reduction in a margin of safety. The removal of some of the excess flow margin, which was introduced by the hot leg streaming flow penalties in later calorimetrics, will allow additional operating margin between the indicated flow and the Technical Specification minimum measured flow limit. The proposed changes in the cold leg elbow tap flow coefficients will continue to be conservative with respect to the analytical model flow predictions, since the proposed coefficients will continue to contain some hot leg streaming penalties from the calorimetric determined coefficients used in the average.

An increase in the RCS flow indication of approximately 1.0% will increase the margin to a reactor trip on low flow but will not adversely affect the plant response to low flow transients. Current UFSAR [updated final safety analysis report] chapter 15 transients that would be expected to cause a reactor trip on the RCS low flow trip setpoint are Partial Loss of Reactor Coolant Flow, Reactor Coolant Pump Shaft Seizure and [RCP] Reactor Coolant Pump Shaft break transients. Three reactor trip functions provide protection for these transients, RCS low flow reactor trip, RCP undervoltage reactor trip and RCP underfrequency reactor trip. The transient analyses of these events assume the reactor is tripped on the low flow reactor trip setpoint. This is conservative and produces a more severe transient response since a reactor trip on undervoltage or underfrequency would normally be expected to trip the reactor sooner and therefore reduce the severity of these transients.

The RCS low flow reactor trip is currently set at 91% of the Technical Specification minimum measured flow of 390,000 gpm. The setpoint will not be revised as a result of this change, which means the transients relying on this function will behave in the same manner with the reactor trips occurring at essentially the same conditions as previously analyzed. Therefore, any small increase in the reactor trip margin gained by the small increase in the indicated RCS flow will not adversely affect the plant response during these low flow events.

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

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

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

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: October 11, 2002.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.9.4, Containment Building Penetrations, to permit the equipment hatch to be open during core alterations and/or during movement of irradiated fuel assemblies within containment. The appropriate TS Bases would also be changed to reflect the proposed changes.

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

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

The proposed changes to the Seabrook Station Technical Specifications (TS) 3.9.4.a, and TS 3.9.4.b do not involve a significant increase in the probability or consequences of an accident previously analyzed. The proposed changes will modify the conditions of containment closure during core alterations or during the movement of irradiated fuel within the containment. Specifically, the proposed changes will permit the new containment outage door to stay open during core alterations or during the movement of irradiated fuel within the containment.

Postulated accidents that could result in a release of radioactive material through the open hatch include a fuel handling accident that results in breaching of the fuel rod cladding, and a loss of residual heat removal (RHR) cooling event that leads to core boiling. The radiological consequences of a design basis fuel handling accident in containment have been evaluated assuming that the containment is open to the outside atmosphere. The calculated offsite and control room doses resulting from a fuel handling accident are less than the criteria specified in USNRC [U.S. Nuclear Regulatory Commission] NUREG–0800, ''Standard Review Plan,'' section 15.7.4 ''Radiological Consequence of Fuel Handling Accident,'' and 10 CFR 50, Appendix A, ''General Design Criteria for Nuclear Power Plants,'' GDC [General Design Criteria]-19, ''Control Room.''

The consequence of a loss of Residual Heat Removal (RHR) is the potential for release of radioactivity outside of containment. Closing containment penetrations is the mitigating action for that consequence. TS 3.9.8.1 and 3.9.8.2 require that corrective actions be taken immediately to restore the RHR cooling as soon as possible if RHR loop requirements are not met (by having one RHR loop operable and in operation). In addition, plant operators are required by the TS to close all containment penetrations providing direct access from the containment atmosphere to the outside environment within 4 hours. Since the most limiting time to boil in this condition (during core alterations or movement of irradiated fuel with at least 23 feet of water above the vessel flange) is approximately 8.3 hours, the risk associated with the potential for the coolant to boil and subsequently cause a release of radioactive gas to the containment atmosphere (if RHR cooling was not restored) is minimal.

The proposed changes to TS 3.9.4.b will add a note pertaining to the personnel hatch airlock within the equipment hatch. The purpose of this note is to provide

clarification that the requirements of TS 3.9.4.b do not apply to the subject personnel hatch airlock when the outage equipment hatch is installed.

Therefore, it is concluded that these proposed [changes] to TS 3.9.4.a and TS 3.9.4.b do not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

The proposed changes to the Seabrook Station Technical Specifications (TS) 3.9.4.a and 3.9.4.b do not create the possibility of a new or different kind of accident from any previously evaluated. The proposed changes will permit the equipment hatch to be open during core alterations and movement of irradiated fuel within the containment building when the containment outage door is installed. The installation of the door does involve a minor change in the present method used to isolate containment penetrations for containment closure. However, the present fuel handling analysis, which is the most limiting event, assumes that the containment is open to the outside atmosphere and the entire airborne radioactivity is instantaneously released to the outside environment. This analysis results in [offsite] doses that are within the guideline values specified in USNRC NUREG–0800, ''Standard Review Plan,'' section 15.7.4 ''Radiological Consequence of Fuel Handling Accident,'' and 10 CFR 50, Appendix A, ''General Design Criteria for Nuclear Power Plants,'' GDC–19, ''Control Room.'' Therefore, the proposed changes to the TS do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes do not involve a significant reduction in [a] margin of safety. The proposed change to TS 3.9.4.a will permit the equipment hatch to be open during core alterations and/or during the movement of irradiated fuel assemblies within containment when the containment outage door is installed and closed or capable of being closed. During movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The calculated offsite and control room operator calculated doses are within the acceptance criteria of USNRC NUREG–0800, ''Standard Review Plan,'' section 15.7.4 ''Radiological Consequence of Fuel Handling Accident,'' and 10 CFR 50, Appendix A, ''General Design Criteria for Nuclear Power Plants,'' GDC–19, ''Control Room.'' Therefore, the proposed changes to TS 3.9.4 do not result in a reduction in [a] margin of safety.

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

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

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

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

Date of amendment request: October 11, 2002.

Description of amendment request: The proposed amendment would change Technical Specification (TS) 3/4.9.3, ''Refueling Operations—Decay Time,'' to revise the time associated with the movement of irradiated fuel in the reactor vessel from 100 hours to 80 hours. The proposed change is based on reanalysis of the radiological consequences of a limiting design basis fuel handling accident using an 80-hour decay time.

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

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

The proposed change to TS 3/4.9.3 does not result in a condition where the design, material, and construction standards that were applicable prior to the proposed change are altered. The probability of occurrence of an accident previously evaluated for Seabrook Station is not altered by the proposed amendment to the technical specifications (TSs). The accidents remain the same as currently analyzed in the Updated Final Safety Analysis Report (UFSAR) as a result of the proposed change to the decay time. The accidents impacted by the new decay time have been reanalyzed and the applicable design limits have not been exceeded. The control room and offsite dose consequences for fuel handling accidents have been reevaluated and continue to meet acceptance limits.

Therefore based on the above discussion, it is concluded that the proposed revision to TS 3/4.9.3 does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed change to the decay time will not create a new accident scenario. The analyses impacted by the revised decay time have been evaluated. The new analysis of the fuel handling accident and spent fuel pool cooling system performance demonstrates that the applicable acceptance criteria continues to be met. The proposed change will not alter the way any structure, system or component functions, and will not significantly alter the manner in which the plant is operated. There will be no significant adverse effect on plant operation or accident mitigation equipment.

Since no new failure modes are created by the proposed revision to TS 3/4.9.3 the proposed change does not create the possibility of a new or different kind of accident from any that was previously evaluated.

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

The fuel handling accident in the fuel building and containment has been reanalyzed for a decay time of 80 hours. The spent fuel pool cooling performance has also been evaluated for the revised decay time. These analyses demonstrate that acceptance criteria are still met for the revised decay time as described herein. The results of the revised analysis show that the resulting offsite doses (based on a decay time period of 80 hours are comparable to the original doses (100-hour decay time period) and well within (< 25%) the limiting values of 10 CFR part 100. Control room doses are also well within the limit of General Design Criteria 19 to 10 CFR part 50, Appendix A. Therefore it is concluded that the proposed decay time still provides sufficient margin to dose consequences from fuel handling and to spent fuel pool temperature limits.

Thus, it is concluded that the proposed revision to TS 3/4.9.3 does not involve a significant reduction in a margin of safety.

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

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

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

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

Date of amendment request: October 11, 2002.

Description of amendment request: The proposed amendment would eliminate the Power Range Neutron Flux High Negative Rate Reactor Trip function from Technical Specification (TS) 3/4.3.1, ''Reactor Trip System Instrumentation,'' TS 2.2.1, ''Reactor Trip System Instrumentation Setpoints,'' and their associated Bases. The proposed changes associated with elimination of the Power Range Neutron Flux High Negative Rate Trip function are based on the NRC-approved analysis provided in Westinghouse WCAP–11394–P–A, ''Methodology for the Analysis of the Dropped Rod Event.'' The proposed amendment would also change TS 3/4.10.3, ''Physics Tests,'' TS

3/4.10.4, ''Reactor Coolant Loops,'' and TS Table 4.3–1, ''Reactor Trip System Instrumentation Surveillance Requirements,'' that are associated with certain testing activities required during STARTUP operations. The proposed changes to TS 3/4.10.3 are to clarify that only the reactor trip Low Setpoint associated with OPERABLE Power Range Neutron Flux instrumentation channels is required to be set at 25% of RATED THERMAL POWER and to reword the time interval for the Analog Channel Operational Test (ACOT) in surveillance requirement (SR) 4.10.3.2 from ''within 12 hours'' to the referenced time interval specified in TS Table 4.3–1, Functional Unit 2.b. In correlation with the proposed change to extend the ACOT interval in SR 4.10.3.2, Table 4.3–1 Note 1, would be changed from ''if not performed in previous 31 days'' to ''if not performed in previous 92 days.'' The proposed change would also extend the ACOT interval for those Functional Units that reference TS Table 4.3–1 Note 1. The proposed change to TS 3/4.10.4 will delete TS 3/4.10.4 in its entirety since the condition allowed by TS 3/4.10.4 (*i.e.*, natural circulation/low flow conditions) was to support the initial startup test program prior to commercial operation. Additionally, as a result of deleting TS 3/4.10.4, the footnote which references TS 3/4.10.4 in TS 3/4.4.1.1 is deleted as well.

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

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

The proposed changes (1) to eliminate the Power Range Neutron Flux High Negative Rate Trip Function, (2) not lowering the Power Range Neutron Flux High Setpoint to the same setpoints as that of the Power Range Neutron Flux Low Setpoint and Intermediate Range reactor trip setpoint prior to conducting Physics Testing, (3) extension of the surveillance interval for performing the ACOT and TADOT [Trip Actuating Device Operational Test] for the above described Reactor Trip System (RTS) Functional Units, (4) elimination of the Special Test Exception allowing performance of Physics Testing under no flow conditions, and (5) the other editorial and Bases changes to support the aforementioned changes do not increase the probability or consequences of reactor core damage accidents resulting from events previously analyzed. The safety functions of other safety related systems and components, which are related to mitigation of these events, have not been altered. All other RTS and Engineered Safety Features Actuation Systems (ESFAS) protection functions are not affected by the proposed changes. Favorable plant-specific historical data as well as industry practice support the proposed change to extend the surveillance intervals for performance of the applicable ACOT or TADOT on the aforementioned instrumentation channels. The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, configuration of the facility, or the manner in which it is operated. The proposed changes do not adversely alter or prevent the ability of structures, systems, or components to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Seabrook Station Updated Final Safety Analysis Report (UFSAR).

Removal of the negative rate trip does not change the probability of a rod drop accident since it does not alter the physical function or characteristic of the rod control system. Changing surveillance intervals for calibrations does not change the probability of an initiating event since historical performance demonstrates that the instrumentation settings will be within the assumed tolerance at the longer interval. Since the effects of the negative rate trip are not considered in the rod drop accident analysis, therefore removal of the trip will not result in an increase in the consequences of the rod drop accident. Changes in surveillance frequencies do not change the essential character of accident progression, thus there is no increase in the consequences.

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

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

The proposed changes do not adversely alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated. No credit is taken in Seabrook Station's safety analyses that is reliant on the Power Range Neutron Flux High Negative Rate Trip Function. Extending the aforementioned surveillance intervals and not lowering the Power Range Neutron Flux High Setpoint prior to physics testing do not create the possibility of a new or different kind of accident from any accident previously evaluated. There are no changes to the source term or radiological release assumptions used in evaluating the radiological consequences in the Seabrook Station UFSAR. The proposed changes have no adverse impact on component or system interactions. The proposed changes will not adversely degrade the ability of systems, structures and components important to safety to perform their safety function nor change the response of any system, structure or component important to safety as described in the UFSAR. The proposed changes do not change the level of programmatic and procedural details of assuring operation of the facility in a safe manner. Since there are no changes to the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated and surveilled, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

There is no adverse impact on equipment design or operation and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. Elimination of the Power Range Neutron Flux High Negative Rate Trip Function will not cause DNB [Departure from Nucleate Boiling] limits to be exceeded since this function is not credited in Seabrook Station's safety analysis. Eliminating the practice of lowering the Power Range Neutron Flux High Setpoint prior to physics testing does not involve a significant reduction in the margin of safety since there is adequate redundancy of nuclear instrumentation channels to prevent core damage from a positive reactivity excursion. The proposed changes to extend certain surveillance intervals do not reduce the reliability of the aforementioned trip functions to operate as designed nor reduce the level of programmatic or procedural controls associated with the aforementioned surveillance requirements. The negative rate trip function could, and has, caused an inadvertent reactor trip. Removal of this function will not reduce any perceived ''defense-in-depth'' since the design of the core limits rod worth such that DNB is acceptable during a rod drop event. Additionally, since WCAP–11394–P–A has demonstrated that the negative rate trip is not considered in the safety analysis margin, removal of the NFRT is not considered a ''significant reduction in margin[.] '' The other changes are editorial/administrative in nature which support the key changes as mentioned above and by their nature do not involve a significant reduction in a margin of safety.

Therefore, the proposed changes as described in this License Amendment Request do not involve a significant reduction in a margin of safety.

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

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

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

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: September 24, 2002.

Description of amendment request: The proposed change will revise Technical Specification (TS) Surveillance Requirement (SR) 4.0.3, to incorporate the approved Consolidated Line Item Improvement Program change associated with the TS Task Force traveler TSTF–358, revision 6, SR 3.0.3, ''Missed Surveillance Requirements.'' Additionally, a change to the Administrative Controls Section, section 6.8, is included in this request to include a new TS requirement for a Bases Control Program, consistent with the Bases Control Program presented in chapter 5, ''Administrative Controls,'' section 5.5, ''Programs and Manuals,'' of the Improved Technical Specifications (ITS) for Westinghouse plants, NUREG 1431, revision 2. 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 model NSHC determination in its application dated September 24, 2002, with the exception of the addition of the incorporation of a Bases Control Program in chapter 5, ''Administrative Control,'' section 5.5, ''Programs and Manuals,'' of the ITS for Westinghouse plants, NUREG 1431, revision 2.

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 for the changes associated with extending the delay period for a missed surveillance 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.

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

SCE&G has reviewed the proposed no significant hazards consideration determination (NSHCD) published in the **Federal Register** as part of the CLIIP [Consolidated Line Item Improvement]. SCE&G has concluded that the proposed NSHCD presented in the **Federal Register** notice is applicable to VCSNS with one exception. The proposed NSHCD is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

The exception is that the published NSHCD does not specifically address the incorporation of a Bases Control Program, as one is already incorporated into the ITS NUREGs. Therefore, a NSHCD is presented for the proposed inclusion of a Bases Control Program into the VCSNS TS.

In accordance with the criteria set forth in 10 CFR 50.92, SCE&G has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided to support this conclusion.

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

The proposed change provides an addition to the Administrative Section of TS to comply with the requirements of the **Federal Register** published notice of availability for TSTF–358, revision 6. This change adds a Bases Control Program to section 6.8 that is consistent with the Bases Control Program in NUREG 1431, revision 2.

A bases control program will not provide for a significant increase in probability or consequences of an accident previously evaluated as there are no changes in hardware or software for the plant and no changes in any operating procedure. The incorporation of a Bases control program into the Administrative Section of TS will help to assure that all assumptions in the plant accident analysis for initial conditions, redundancy, and independence are maintained. This change will assure that any and all future revisions to the Bases section of TS will be consistently controlled in a manner acceptable to both the industry and the NRC.

Therefore, this change provides for no 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 has no impact on the operation of the plant or changes to plant configuration. Only the manner in which VCSNS processes and distributes a TS Bases change will be revised and the controls will be similar to the majority of the industry. The NRC has approved the methodology used in the Bases control program, located in section 5.5 of the Westinghouse Standardized Technical Specifications, NUREG 1431, revision 2.

Therefore, there is no possibility of this change creating a new or different kind of accident from any previously evaluated.

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

This change provided for a standardized methodology, acceptable to the NRC, to assure consistent guidance for Bases changes is provided and the process is controlled under a TS administrative program. No impact to any plant hardware or safety analysis will occur from this proposed change. Therefore, there is no significant reduction in any margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: John A. Nakoski.

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

Date of amendment request: November 6, 2002.

Description of amendment request: The proposed amendment would revise the Browns Ferry Nuclear Plant (BFN), Units 2 and 3, Reactor Pressure Vessel (RPV) material surveillance program required by 10 CFR 50, Appendix H. This program incorporates the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) into the BFN Units 2 and 3 licensing basis.

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

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

The proposed change implements a [an] integrated surveillance program that has been evaluated by the NRC staff as meeting the requirements of paragraph III.C of Appendix H to 10 CFR 50. Consequently, the change does not significantly increase the probability of any accident previously evaluated. The change provides the same assurance of RPV integrity. The change will not cause the reactor pressure vessel or interfacing systems to be operated outside their design or testing limits. Also, the change will not alter any assumptions previously made in evaluating the radiological consequences of accidents. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change revises the BFN Units 2 and 3 licensing basis to reflect participation in the BWRVIP ISP. The proposed change does not involve a modification of the design of plant structures, systems, or components. The change will not impact the manner in which the plant is operated as plant operating and testing procedures will not be affected by the change. The change will not degrade the reliability of structures, systems, or components important to safety as equipment protection features will not be deleted or modified, equipment redundancy or independence will not be reduced, supporting system performance will not be increased, and increased or more severe testing of equipment will not be imposed. No new accident types or failure modes will be introduced as a result of this proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from that previously evaluated.

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

The proposed change has been evaluated as providing an acceptable alternative to the plant specific RPV material surveillance program and meets the requirements of 10 CFR 50 Appendix H for RPV material surveillance.

Appendix G to 10 CFR 50 describes the conditions that require pressure temperature (P/T) limits and provides the general bases for these limits. Until the results from the Integrated Surveillance Program become available, RG [Regulatory Guide] 1.99, revision 2 will be used to predict the amount of neutron irradiation damage. The use of operating limits based on these criteria, as defined by applicable regulations, codes, and standards, provide reasonable assurance that nonductile or rapidly propagating failure will not occur. The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the reactor coolant pressure boundary (RCPB). Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

The proposed change will not affect any safety limits, limiting safety system settings, or limiting conditions of operation. The proposed change does not represent a change in initial conditions, or in a system response time, or in any other parameter affecting the course of an accident analysis supporting the Bases of any Technical Specification. Further, the proposed change does not involve a revision to P/T limits but rather a revision to the surveillance capsule withdrawal schedule for the second surveillance capsule. The current P/T limits were established based on adjusted reference temperatures for RPV beltline materials calculated in accordance with RG 1.99, revision 2. P/T limits will continue to be revised, as necessary, for changes in adjusted reference temperature due to changes in fluence when two or more credible surveillance data sets become available. When two or more credible surveillance data sets become available, P/T limits will be revised as prescribed by RG 1.99, revision 2 or other NRC approved guidance. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Allen G. Howe.

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

Date of application request: October 3, 2002.

Description of amendment request: The amendment would revise Tables 3.3.1–1 (Reactor Trip System (RTS) Instrumentation) and 3.3.2–1 (Engineered Safety Feature Actuation System (ESFAS) Instrumentation) of Limiting Conditions for Operation (LCO) 3.3.1, ''RTS Instrumentation,'' and 3.3.2, ''ESFAS Instrumentation,'' of the Technical Specifications. The proposed changes are to the steam generator (SG) water level low-low (adverse and normal containment environment) functions.

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

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

Overall protection system performance [for the proposed changes] will remain within the bounds of the previously performed accident analyses since there are no hardware changes. The design of the SG water level sensing equipment and the coincidence logic in the Solid State Protection System will be unaffected. The only physical change to the RTS and ESFAS instrumentation is the increased actuation setpoints in the NAL

bistable comparator cards in the 7300 Process Protection System. These changes have already been implemented in the field and are in the conservative direction, *i.e.*, a trip actuation signal will be generated sooner for an event that challenges the ability of the steam generators to provide a heat sink. In all other regards, the design of the RTS and ESFAS instrumentation will be unaffected. These 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 this amendment request are maintained.

The probability and consequences of accidents previously evaluated in the FSAR [Callaway Final Safety Analysis Report] are not adversely affected because the changes to the RTS and ESFAS trip setpoints assure the conservative response of the affected trip functions, consistent with the safety analysis and licensing basis.

The proposed changes will not affect the probability of any event initiators. 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 changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

There are no hardware changes, other than increased bistable setpoints in the adjustable bistable comparator cards that have already been implemented, 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 parameters. The LCO Applicability exception for the SG Water Level Low-Low (Normal Containment Environment) channels recognizes the functional design of the system that enables the SG Water Level Low-Low (Adverse Containment Environment) channels with a higher water level trip setpoint whenever the Containment Pressure—Environmental Allowance Modifier channels in the same protection sets are tripped. No performance requirements or response time limits will be affected.

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

The proposed changes do not eliminate any RTS surveillance or alter the frequency of surveillances required by the Technical Specifications. The nominal Trip Setpoints specified in the Technical Specification Bases have already been increased in the conservative direction. The safety analysis limits assumed in the transient and accident analyses are unchanged. None of the acceptance criteria for any accident analysis are 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: October 3, 2002.

Description of amendment request: The amendment would add a phrase to Limiting Condition for Operation (LCO) 3.1.8, ''Physics Tests Exceptions—Mode 2,'' of the Technical Specifications. The phrase to be added is that the number of required channels for certain functions in Table 3.3.1–1 of LCO 3.3.1, ''RTS Instrumentation,'' may be reduced from four to three required channels. LCO 3.1.8 applies to reactor Mode 2 during physics tests.

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 [for the proposed change] will remain within the bounds of the previously performed accident analyses since there are no permanent hardware changes. The design of the RTS [reactor trip system] instrumentation will be unaffected; only the manner in which the system is connected for short duration physics testing is being changed to allow the temporary bypass of one power range channel. The reactor protection system will continue to function in a manner consistent with the plant design basis since a sufficient number of power range channels will remain OPERABLE to assure the capability of protective functions, even with a postulated single failure. [The number of required channels for certain functions in Table 3.3.1–1 is only being reduced from 4 to 3 channels.] All design, material, and construction standards that were applicable prior to the request are maintained.

The proposed change will allow the temporary bypass of one power range neutron flux channel during the performance of low power physics testing in MODE 2. This results in a temporary change to the coincidence logic from one-out-of-three under the current TS (with a trip imposed on the channel used for physics testing) to two-out-of-three under the proposed TS (the channel used for physics testing would be in a bypassed state). However, this two-out-of-three coincidence logic still supports [the] required protection and control system applications, while reducing plant susceptibility to a spurious reactor trip.

The proposed change will not affect the probability of any event initiators. There will be no change to normal plant operating parameters or accident mitigation performance.

The proposed change will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR [Callaway Final Safety Analysis Report].

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

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

There are no permanent hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. This change will not affect the normal method of power operation or change any operating parameters. No performance requirements will be affected.

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.

The proposed amendment does not alter the design or performance of the 7300 Process Protection System, Nuclear

Instrumentation System (other than as discussed above), or Solid State Protection System used in the plant protection systems. [The number of the required channels is not an initiator of an accident.]

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.

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 (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.

The proposed change does not eliminate any RTS surveillance or alter the Frequency of surveillances required by the Technical Specifications. The nominal RTS and Engineered Safety Features Actuation System (ESFAS) trip setpoints (TS Bases Tables B 3.3.1–1 and B 3.3.2–1), RTS and ESFAS allowable values (TS 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. The potential reduction in the frequency of spurious reactor trips would effectively increase the margin of safety or, at a minium, be risk-neutral.

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

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

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

NRC Section Chief: Stephen Dembek.

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

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the **Federal Register** on the day and page cited. This notice does not extend the notice period of the original notice.

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

Date of amendment request: October 3, 2002.

Brief description of amendment request: The proposed amendment would revise the definition of steam generator (SG) tube inspection in Technical Specification 5.5.9, ''Steam Generator Tube Surveillance Program.'' The amendment would add a requirement for using the rotating pancake coil (RPC) to the H* depth in the tubesheet. The proposed amendment is based on the Westinghouse Topical Report WCAP–15932–P, ''Improved Justification of Partial-Length RPC Inspection of Tube Joints of Model F Steam Generators of Ameren-UE Callaway Plant,'' revision 0, dated September 2002.

*Date of publication of individual notice in **Federal Register:*** October 18, 2002 (67 FR 64422).

Expiration date of individual notice: November 18, 2002.

Notice of Issuance of Amendments to Facility Operating Licenses

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

Notice of consideration of issuance of amendment to facility operating license, proposed no significant hazards consideration determination, and opportunity for a Hearing in connection with these actions was published in the **Federal Register** as indicated.

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

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

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2), Darlington County, South Carolina

Date of application for amendment: May 6, 2002, as supplemented July 25, August 12, September 6, October 15, and October 31, 2002.

Brief description of amendment: This amendment increases the HBRSEP2 maximum steady-state core power level from 2300 megawatts thermal (MWt) to 2339 MWt, an increase of approximately 1.7 percent.

Date of issuance: November 5, 2002.

Effective date: November 5, 2002.

Amendment No.: 196.

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

*Date of initial notice in **Federal Register:*** September 3, 2002 (67 FR 56319). The July 25, August 12, September 6, October 15, and October 31, 2002, supplements contained clarifying information only and did not change the initial no significant hazards consideration determination or expand the scope of the initial application.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 14, 2002, as supplemented by letter dated September 9, 2002.

Brief description of amendment: The amendment revised Surveillance Requirement (SR) 4.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period was extended from the current limit of ''* * * up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours'' to ''* * * up to 24 hours or up to the limit of the specified interval, whichever is greater.'' In addition, the following requirement was added to SR 4.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.'' Also, a Bases Control Program was added as Technical Specification 6.5.14, clarifications were made to SR 4.0.1, and other minor changes were made to SR 4.0.3, consistent with NUREG–1432, revision 2, ''Standard Technical Specifications, Combustion Engineering Plants.''

Date of issuance: November 1, 2002.

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

Amendment No.: 246.

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

Date of initial notice in ***Federal Register:*** July 23, 2002 (67 FR 48216). The application was renoticed on October 1, 2002 (67 FR 61680).

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

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: July 9, 2002.

Brief description of amendment: The amendment revised Technical Specification Sections 3.10.4, ''Rod Insertion Limits,'' 3.10.5, ''Rod Misalignment Limitations,'' and 3.10.6, ''Inoperable Rod Position Indicator Channels,'' to remove the cycle-specific allowances on (1) rod insertion limits during individual rod position indicator channel calibrations and (2) rod position indicator channel accuracy for operation at or below 50 percent power. The amendment also revises the control rod indicated misalignment limits.

Date of issuance: November 7, 2002.

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

Amendment No.: 234.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** October 7, 2002 (67 FR 62500).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 8, 2002.

Brief description of amendments: The proposed amendments would change Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF–11 and NPF–18. Specifically, the proposed change adds two footnotes to TS Table 3.3.8.1–1, ''Loss of Power Instrumentation,'' Functions 1.e and 2.e, ''Degraded Voltage—Time Delay, LOCA,'' and makes an editorial change to the heading of TS Table 3.3.8.1–1. The Degraded Voltage—Time Delay, LOCA, function is currently required to be OPERABLE during plant configurations when the ECCS instrumentation that generates the Loss of Coolant Accident (LOCA) signal is not required to be OPERABLE. The proposed changes correct this inconsistency by adding two new footnotes to TS Table 3.3.8.1-i that modify the required OPERABILITY of the Degraded Voltage—Time Delay, LOCA, function.

Date of issuance: November 12, 2002.

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

Amendment Nos.: 155 & 141.

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 20, 2002 (67 FR 53986).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 12, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 1, 2002, as supplemented October 23, 2002.

Brief description of amendments: The amendments revise the licensing basis as described in the Updated Final Safety Analysis Report to allow lifting heavier loads with the reactor building crane during the Unit 1 refueling outage beginning in November 2002.

Date of issuance: November 4, 2002.

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

Amendment Nos.: 209 & 204.

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

Date of initial notice in ***Federal Register:*** October 4, 2002 (67 FR 62270)

The supplement dated October 23, 2002, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket No. 50–254, Quad Cities Nuclear Power Station, Unit 1, Rock Island County, Illinois

Date of application for amendment: May 30, 2002, as supplemented August 15 and October 18, 2002.

Brief description of amendment: The amendment revises the safety limit minimum critical power ratio for two-loop and single-loop operation for Unit 1 for Cycle 18.

Date of issuance: November 14, 2002.

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

Amendment No.: 210.

Facility Operating License No. DPR–29: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** July 9, 2002 (67 FR 45569).

The supplements dated August 15 and October 18, 2002, provided additional information that clarified the application, did not change the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 21, 2001, as supplemented January 25, 2002, and August 15, 2002.

Brief description of amendments: The amendments revised the Technical Specifications (TS) Surveillance Requirement (SR) 4.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period is extended from ''* * * up to 24 hours to permit completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours' to ''* * * up to 24 hours or up to the limit of the specified frequency, whichever is greater.'' In addition, the following requirement was added to SR 4.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.'' Lastly, an editorial change moved two sentences dealing with operability requirements from SR 4.0.3 to SR 4.0.1 to make the revised TS consistent with the Standard TS for Combustion Engineering plants.

Date of Issuance: November 4, 2002.

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

Amendment Nos.: 186 and 129.

Facility Operating License Nos. DPR–67 and NPF–16: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** September 17, 2002 (67 FR 58645).

The January 25, 2002, and August 15, 2002, Supplements did not affect the original proposed no significant hazards determination, or expand the scope of the request as noticed in the ***Federal Register***.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 28, 2002.

Brief description of amendment: The amendment revised the Technical Specifications. Specifically, it revised item 9, Shutdown Cooling System Isolation High Area Temperature, of Table 4.6.2b, ''Instrumentation that Initiates Primary Coolant System or Containment Isolation,'' changing the frequency of instrument channel test and instrument channel calibration from ''once during each major refueling outage'' to ''once per operating cycle.''

Date of issuance: November 13, 2002.

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

Amendment No.: 177.

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

Date of initial notice in ***Federal Register:*** August 6, 2002 (67 FR 50956).

The staff's related evaluation of the amendment is contained in a Safety Evaluation dated November 13, 2002.

No significant hazards consideration comments received: No.

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

Date of amendment request: May 17, 2002, as supplemented on June 28, July 1, August 29, and October 11, 2002.

Description of amendment request: The amendment revises the license to reflect changes related to the transfer of the license for Seabrook Station, Unit No. 1, previously held by North Atlantic Energy Service Corporation (NAESCO), as the licensed operator of the facility, and certain co-owners of the facility, on whose behalf NAESCO is also acting, to FPL Energy Seabrook, LLC.

Date of issuance: November 1, 2002.

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

Amendment No.: 86.

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

Date of initial notice in ***Federal Register:*** June 14, 2002 (67 FR 40972).

The letters dated June 28, July 1, July 24, August 29, and October 11, 2002, provided clarifying information and did not expand the application beyond the scope of the notice or affect the applicability of the Commission's generic no significant hazards consideration determination pursuant to 10 CFR 2.1315.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 25, 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: July 12, 2002.

Brief description of amendment: The amendment revised the Kewaunee Nuclear Power Plant Technical Specification (TS) 3.1.a.3, ''Pressurizer Safety Valves'' to make it consistent with the Improved Standard TS to improve clarity. The amendment allows both pressurizer safety valves to be inoperable or removed while the reactor vessel head is on, provided the reactor coolant system (RCS) cold legs temperature is below 200 degrees F, which is in MODE 5 configuration. During MODE 5 configuration, the low temperature over pressure protection system is available and operable to protect the RCS from overpressure.

Date of issuance: November 7, 2002.

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

Amendment No.: 164.

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

Date of initial notice in ***Federal Register:*** August 6, 2002 (67 FR 50957).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 27, 2001, as supplemented by letter dated August 12, 2002.

Brief description of amendments: The amendments delete Section 6.8.4.e, ''Post-Accident Sampling,'' from the Salem Nuclear Generating Station, Unit Nos. 1 and 2, Technical Specifications, and License Condition 2.C.25, ''Post-Accident Sampling,'' for Unit 2, thereby eliminating the requirements to have and maintain the post-accident sampling program.

Date of issuance: November 5, 2002.

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

Amendment Nos.: 254 and 235.

Facility Operating License Nos. DPR–70 and DPR–75: The amendments revised the Technical Specifications and License.

Date of initial notice in ***Federal Register:*** October 31, 2001 (66 FR 55022).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 5, 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 8, 2002.

Brief description of amendment: This amendment changes TS 3.7.6 to exclude the control room normal and emergency air handling system from having to include TS 3.0.4 requirements when applying the action requirements of Limiting Condition for Operation 3.7.6 in Modes 5 and 6. Specifically, the change will allow operation in a manner that is already permitted by TS 3.7.6.

Date of issuance: November 7, 2002.

Effective date: November 7, 2002.

Amendment No.: 161.

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

Date of initial notice in ***Federal Register:*** June 25, 2002 (67 FR 42829).

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

No significant hazards consideration comments received: No.

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

Date of amendment request: July 25, 2002, as supplemented by letter dated August 30, 2002.

Brief description of amendment: The amendment revises paragraphs in Section 5.0, ''Administrative Controls,'' of the Technical Specifications to allow the use of generic personnel titles in place of plant-specific personnel titles.

Date of issuance: November 6, 2002.

Effective date: November 6, 2002, and shall be implemented within 30 days of the date of issuance including the approval of the Updated Safety Analysis Report (USAR) change request that incorporates the relationships between the titles in ANSI/ANS–3.1–1978 and the plant-specific personnel titles in the USAR, as described in the licensee's letters of July 25 and August 30, 2002.

Amendment No.: 149.

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

Date of initial notice in ***Federal Register:*** August 20, 2002 (67 FR 53993).

The August 30, 2002, supplemental letter provided additional information that clarified the application, did not change the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 6, 2002.

No significant hazards consideration comments received: No.

Dated in Rockville, Maryland, this 18th day of November 2002.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. 02–29737 Filed 11–25–02; 8:45 am]

BILLING CODE 7590–01–P

PENSION BENEFIT GUARANTY CORPORATION

Submission of Information Collection for OMB Review; Comment Request; Notice of Failure To Make Required Contributions

AGENCY: Pension Benefit Guaranty Corporation.

ACTION: Notice of request for extension of OMB approval.

SUMMARY: The Pension Benefit Guaranty Corporation (PBGC) is requesting that the Office of Management and Budget (OMB) extend approval, under the Paperwork Reduction Act, of the collection of information under Part 4043 of its regulations relating to Notice of Failure to Make Required Contributions (OMB control number 1212–0041; expires January 31, 2003). This notice informs the public of the PBGC's request and solicits public comment on the collection of information.

DATES: Comments should be submitted by December 26, 2002.

ADDRESSES: Comments should be mailed to the Office of Information and Regulatory Affairs of the Office of Management and Budget, Attention: Desk Officer for Pension Benefit Guaranty Corporation, Washington, DC 20503.

Copies of the request for extension (including the collection of information) may be obtained without charge by writing to the PBGC's Communications and Public Affairs Department, suite 240, 1200 K Street, NW., Washington, DC 20005–4026, or by visiting that office or calling 202–326–4040 during normal business hours. (TTY and TDD users may call the Federal relay service toll-free at 1–800–877–8339 and ask to be connected to 202–326–4040.) The regulations, forms, and instructions relating to the notice of failure to make required contributions may be accessed on the PBGC's Web site at *http://www.pbgc.gov.*

FOR FURTHER INFORMATION CONTACT: James L. Beller, Attorney, Office of the General Counsel, Pension Benefit Guaranty Corporation, 1200 K Street, NW., Washington, DC 20005–4026, 202–326–4024. (TTY and TDD users may call the Federal relay service toll-free at 1–800–877–8339 and ask to be connected to 202–326–4024.)

SUPPLEMENTARY INFORMATION: Section 302(f) of the Employee Retirement Income Security Act of 1974 (''ERISA'') and section 412(n) of the Internal Revenue Code of 1986 (''Code'') impose a lien in favor of an underfunded single-employer plan that is covered by the termination insurance program if (1) any person fails to make a required payment when due, and (2) the unpaid balance of that payment (including interest), when added to the aggregate unpaid balance of all preceding payments for which payment was not made when due (including interest), exceeds $1 million. (For this purpose, a plan is underfunded if its funded current liability percentage is less than 100 percent.) The lien is upon all property and rights to property belonging to the person or persons who are liable for required contributions (*i.e.*, a contributing sponsor and each member of the controlled group of which that contributing sponsor is a member).

Only the PBGC (or, at its direction, the plan's contributing sponsor or a member of the same controlled group) may perfect and enforce this lien. Therefore, ERISA and the Code require persons committing payment failures to notify the PBGC within 10 days of the due date whenever there is a failure to make a required payment and the total of the unpaid balances (including interest) exceeds $1 million.

PBGC Form 200, Notice of Failure to Make Required Contributions, and related filing instructions, implement the statutory notification requirement. Submission of Form 200 is required by 29 CFR § 4043.81.

The collection of information under the regulation has been approved through January 31, 2003, by OMB under control number 1212–0041. The PBGC is requesting that OMB extend approval for another three years. An agency may not conduct or sponsor, and