limited; however, the meeting site is located adjacent to the White Flint Station on the Metro Red Line.

FOR FURTHER INFORMATION CONTACT: Theodore S. Sherr, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone: (301) 415–7190, e-mail tss@nrc.gov.

Dated at Rockville, Maryland this 16th day of March, 2000.

For the Nuclear Regulatory Commission.

Theodore S. Sherr,

Chief, Licensing and International Safeguards Branch, Division of Fuel Cycle Safety and Safeguards, NMSS.

[FR Doc. 00–7102 Filed 3–21–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from February 26, 2000, through March 10, 2000. The last biweekly notice was published on March 8, 2000 (65 FR 12286).

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 Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By April 21, 2000, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). 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: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

Commonwealth Edison Company, Docket No. 50–237, Dresden Nuclear Power Station, Unit 2, Grundy County, Illinois

Date of amendment request: April 30, 1999.

Description of amendment request: The proposed amendment would revise the expiration date of the operating license to allow 40 years of operation from the original date of issuance.

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:

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

The programs to detect incipient failures or degraded performance such as Inservice Inspection, Inservice Testing, and Environmental Qualification programs, for example, remain in place and unchanged. The thermal cycles and reactor vessel toughness are within the 40-year design margin and will remain within those margins for the total operating period proposed by the amendment. No equipment is added, modified, or removed as a result of this amendment. Therefore there is no increase in the probability of an occurrence. No changes are made to the assumptions on which the UFSAR accident and transient analyses are based. Therefore, there is no reason for an increase in the consequences of any of the analyzed conditions which could lead to an increase in Onsite or Offsite dose consequences.

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

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

No systems, structures, or components are changed by this amendment. No procedures that operate, maintain, or surveil them are changed. No provisions of the license or the technical specifications are modified or relaxed.

Therefore, the proposed amendment does nor create the possibility of a new or different kind of accident from any previously evaluated.

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

No assumptions are changed for any analysis as a result of this amendment. No system, structure, or component is changed by this amendment. This amendment does not change the results of accident and transient analyses previously evaluated.

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

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

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of amendment request: February 21, 2000.

Description of amendment request: The proposed amendments would change the condensate storage tank (CST) low level setpoint to prevent entrainment of air in the high pressure coolant injection (HPCI) pump suction line when taking suction from the CST. The amendments would also revise the surveillance requirements for the CST level instruments.

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:

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

The Condensate Storage Tank (CST) water level and the installation of new pressure type switches are not precursors to accidents or transients described in the Updated Final Safety Analysis Report (UFSAR). The proposed changes will maintain the operability of the High Pressure Coolant Injection (HPCI) system, thus the HPCI system will continue to function as designed. Any failure of the new switches will still cause realignment of the HPCI suction from the CST to the Torus as currently designed. Therefore, the proposed changes in water level and the installation of a new type switch will not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

For a system to create the possibility of a new and different accident, the proposed changes would have to require the system to operate in a mode or configuration that is different from the original design. The installation of the new switches does not alter the current logic configuration. The new switches will continue to function and initiate a transfer from the CSTs to the Torus as the suction source as originally designed. The proposed changes to the Technical Specifications (TS) will ensure that the HPCI suction transfer will occur before any air is entrained into the pump suction line. This is accomplished by ensuring that the water level in the CSTs does not reach the vortex limit before the transfer of the HPCI pump suction from the CSTs to the Torus is complete. No new functional failure modes will be introduced upon implementation of the proposed changes. Therefore, the possibility of a new or different kind of accident has not been created.

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

The proposed changes to the CST Level-Low trip setpoint and installation of the new pressure switches provide assurance that air entrainment and vortexing will be prevented during HPCI operation. By maintaining an increased volume in the CSTs, the probability of a HPCI system malfunction due to air entrainment or vortexing is decreased. The installation of the new pressure type switches does not change the current logic configuration. The new switches will be calibrated at a frequency to ensure that the probability of unacceptable instrument drift is maintained at an acceptable level. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

Commonwealth Edison Company, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: February 28, 2000.

Description of amendment request: The proposed amendments would increase the Technical Specification safety limit for the Minimum Critical Power Ratio from 1.08 for two loop operation and 1.09 for single loop operation to 1.11 and 1.12 respectively. The revised safety limits will conservatively bound the current LaSalle Unit 2 operating cycle for an anticipated 5 percent power uprate.

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

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

The proposed changes increase the two loop operation Minimum Critical Power Ratio (MCPR) Safety Limit from 1.08 to 1.11 and the single loop operation MCPR Safety Limit from 1.09 to 1.12. MCPR Safety Limits have been established consistent with NRC-approved methods to ensure that fuel performance is acceptable. These changes do not affect the operability of plant systems, nor do they compromise any fuel performance limits. Therefore, the probability of an accident will not be changed based on these proposed changes.

The MCPR Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. A larger value for the MCPR Safety Limit is conservative and bounding for the current LaSalle County Station, Unit 2, Cycle 8 core at the current licensed power level, because compliance with an MCPR Safety Limit equal to or greater than the calculated value will ensure that less than 0.1% of the fuel rods experience boiling transition. The MCPR Safety Limit does not impact the source term or pathways assumed in accidents previously evaluated. Therefore, these proposed changes do not increase the consequences of an accident previously evaluated.

Additionally, operational MCPR limits will be applied that will ensure the MCPR Safety Limit is not violated during all modes of operation and anticipated operational occurrences in accordance with the Core Operating Limits Report (COLR), which will be implemented prior to operation at uprated power. The MCPR Safety Limit ensures that less than 0.1% of the fuel rods in the core are expected to experience boiling transition. Therefore, the probability or consequences of an accident will not increase.

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

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. Changing the MCPR Safety Limit does not alter or add any new equipment or change modes of operation. The MCPR Safety Limit is established to ensure that 99.9% of the fuel rods avoid boiling transition.

The MCPR Safety Limit is changing for LaSalle County Station, Unit 2 to support Cycle 8 operation at uprated power conditions. Changing the MCPR Safety Limit does not introduce any physical changes to the plant, alter the processes used to operate the plant, or change allowable modes of operation. Therefore, no new or different kind of accident is created that is different from any accident previously evaluated.

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

The MCPR Safety Limit provides a margin of safety by ensuring that less than 0.1% of the fuel rods are predicted to be in boiling transition. The proposed changes increase the two loop operation MCPR Safety Limit from 1.08 to 1.11 and the single loop operation MCPR Safety Limit from 1.09 to 1.12. A larger value for the MCPR Safety Limit is conservative and bounding for the current LaSalle County Station, Unit 2 Cycle 8 core at the current licensed power level, because compliance with a MCPR Safety Limit equal to or greater than what is calculated will ensure that less than 0.1% of the fuel rods experience boiling transition. Additionally, the proposed changes are being submitted prior to completion of the detailed calculations for Cycle 8 power uprate. However, based on preliminary calculations, these revised limits are anticipated to bound Unit 2 Cycle 8 operation at uprated conditions.

Therefore, the margin of safety will not be reduced.

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

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: November 23, 1999.

Description of amendment request: The proposed amendments would revise Technical Specification 5.5.11—Ventilation Filter Testing Program, which provides the test requirements for charcoal filters, to assure compliance with the requirements of American Society for Testing and Materials (ASTM) D3803–1989.

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

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

No. The proposed changes will ensure that the Technical Specification 5.5.11, Section c, required testing of charcoal filters in McGuire ventilation systems designed to meet the guidance provided in Regulatory Guide 1.52, Revision 2, are performed as per ASTM D3803–1989. This will ensure that these filters are capable of performing their design function to maintain offsite and control room operator doses within the limits of 10 CFR 100, Subpart A and 10 CFR 50, Appendix A, GDC [General Design Criteria] 19, following a LOCA [Loss-of-Coolant Accident] or a postulated fuel handling accident. Consequently, the proposed changes only deal with the performance of these systems during an accident and have no impact on accident probabilities. In addition, since the proposed changes help ensure the capability of the subject ventilation systems to perform their design function, there will be no reduction in the ability of these systems to minimize the consequences of a previously evaluated accident.

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

No. The proposed changes only help ensure the performance of the subject ventilation systems during an accident and have no impact on accident possibilities. No changes are being made to actual plant hardware or the way in which the plant is being operated. Therefore, no new accident causal mechanisms will be generated. Consequently, plant accident analyses will not be affected by these changes.

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

No. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following accident conditions. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these barriers will not be degraded by the proposed changes. In addition, the proposed changes to the maximum methyl iodide requirements to accommodate planned changes in filter efficiencies will not result in any degradation in the capability of the affected charcoal filters to perform their design function. As a result of the above, plant safety analyses will not be affected by the changes proposed in this LAR [License Amendment Request].

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

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

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: January 6, 2000.

Description of amendment request: The proposed amendments would revise Technical Specifications (TS) 3.3.1—Reactor Trip System (RTS) Instrumentation, TS 3.3.2—Engineered Safety Feature Actuation System (ESFAS) Instrumentation, TS 3.3.5—Loss of Power Diesel Generator Start (LOP) Instrumentation, and TS 3.3.6—Containment Purge and Exhaust Isolation (VP) Instrumentation. The proposed revisions will facilitate treatment of the applicable RTS, ESFAS, LOP, and VP Instrumentation TS Trip Setpoints as nominal values. In addition, proposed changes to the applicable TS Bases further define the TS Trip Setpoints as nominal values.

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

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

No. The proposed changes are consistent with the current licensing basis for the McGuire Nuclear Station, the setpoint methodologies used to develop the Trip Setpoints, the McGuire Safety Analyses, and current station calibration procedures and practices. The Reactor Trip System and Engineered Safety Features Actuation System are not accident initiating systems; they are accident mitigating systems. Therefore, these proposed changes will have no impact on any accident probabilities. Accident consequences will not be affected, as no changes are being made to the plant which will involve a reduction in reliability of these systems. Consequently, any previous evaluations associated with accidents will not be affected by these changes.

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

No. The proposed changes are consistent with the current licensing basis for the McGuire Nuclear Station, the setpoint methodologies used to develop the Trip Setpoints, the McGuire Safety Analyses, and current station calibration procedures and practices. No changes are being made to actual plant hardware which will result in any new accident causal mechanisms. Also, no changes are being made to the way in which the plant is being operated. Therefore, no new accident causal mechanisms will be generated. Consequently, plant accident analyses will not be affected by these changes.

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

No. The proposed changes are consistent with the current licensing basis for the McGuire Nuclear Station, the setpoint methodologies used to develop the Trip Setpoints, the McGuire Safety Analyses, and current station calibration procedures and practices. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following accident conditions. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these barriers will not be degraded by the proposed changes. Consequently, plant safety analyses will not be affected by these changes.

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, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: Richard L. Emch, Jr.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: November 23, 1999, as supplemented by letter dated February 24, 2000

Description of amendment request: The proposed amendment would incorporate the use of American Society for Testing and Materials (ASTM) D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon,'' into the Technical Specifications (TSs). Entergy Operations, Inc. (the licensee) is submitting this proposed amendment as a complete response to Nuclear Regulatory Commission (NRC) Generic Letter (GL) 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.'' The February 24, 2000, supplement proposes additional changes to the TSs to ensure that ventilation system velocity requirements are established in accordance with the standards of ASTM D3803–1989. This application was previously noticed in the **Federal Register** on March 8, 2000 (65 FR 12291).

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:

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

Deleting portions of applicable ANO–1 [Arkansas Nuclear One, Unit 1] TSs that reference system design velocity criteria for activated charcoal medium testing requires no physical change to plant design. NRC GL 99–02, in support of the ASTM D3803–1989 standard, requires licensees to utilize charcoal testing methods that will ensure the current license basis, as it relates to General Design Criterion (GDC) 19, is maintained. The existing criterion within the affected ANO–1 TSs is less restrictive than that of ASTM D3803–1989 standard and, therefore, is being proposed for deletion. The testing of charcoal mediums has no impact on the probability of an accident occuring. However, the charcoal mediums do act to reduce radioiodines released to the environment during and following an accident. Testing the charcoal mediums to a more restrictive standard, however, does not increase the consequences of an accident since such testing ensures the current analyses remain valid.

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

Criterion 2—Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated

As stated previously, the proposed changes to the ANO–1 TSs do not result in any physical change to plant design, nor does the testing of charcoal mediums act to create a new or different accident than that previously analyzed. The existing criterion within the affected ANO–1 TSs is less restrictive than that of ASTM D3803–1989 standard and, therefore, is being proposed for deletion. Testing criteria governing the operability of charcoal mediums is not considered an accident initiator of new, different, or previously analyzed accidents. The charcoal mediums act solely to reduce radioiodines released to the environment during and following accident scenarios.

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

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

Testing of charcoal mediums to more restrictive criteria acts to better ensure that these mediums will perform their design function during and following accidents that result in a release of radioiodines. No reduction in the margin to safety can be construed based on the new testing criteria. The charcoal mediums will continue to remove radioiodines as originally designed and approved by the NRC during and following accidents involving radioactive release.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: January 27, 2000.

Description of amendment request: The proposed amendment would revise the Arkansas Nuclear One, Unit 2 (ANO–2) technical specifications (TS) by providing actions associated with inoperable control room emergency ventilation or cooling systems during movement of irradiated fuel during shutdown modes of operation, when allowed outage times associated with these systems are not met.

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:

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

The inclusion of additional actions within the ANO–2 TSs associated with the control room emergency ventilation and air conditioning systems during the handling of irradiated fuel does not require any physical modification to plant components or systems. Implementing the proposed actions act to ensure the operability of the remaining system, eliminate the reliance on automatic actuation where applicable, and ensure that any active failure will be readily detected. The proposed changes, therefore, act to ensure [that] the consequences of a fuel handling accident are mitigated and have no impact on the probability [of] a fuel handling accident occurring. The proposed actions are in addition to those currently required by the ANO–2 TSs and, therefore, are more restrictive.

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

Criterion 2—Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated

The inclusion of additional actions within the ANO–2 TSs associated with the control room emergency ventilation and air conditioning systems during the handling of irradiated fuel does not require any physical modification to plant components or systems. Implementing the proposed actions act to ensure the operability of the remaining system, eliminate the reliance on automatic actuation where applicable, and ensure that any active failure will be readily detected. The proposed changes, therefore, are not relevant to creating new or different kinds of accidents than those previously evaluated. The proposed actions are in addition to those currently required by the ANO–2 TSs.

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

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

The inclusion of additional actions within the ANO–2 TSs associated with the control room emergency ventilation and air conditioning systems during the handling of irradiated fuel act to ensure the operability of the remaining system, eliminate the reliance on automatic actuation where applicable, and ensure that any active failure will be readily detected. The proposed changes, therefore, act to maintain the margin of safety by ensuring the operability of redundant equipment that is required to protect control room personnel in the event of a fuel handling accident. The proposed actions are in addition to those currently required by the ANO–2 TSs and, therefore, are more restrictive.

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

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

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801.

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: February 24, 2000.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 4.4.11 on reactor coolant system vent flow verification, TS 4.6.1.1.a on containment penetration closure verification (non-automatic), and TS 4.6.3.1.2 on containment isolation valve actuation verification. These TS surveillances require testing to be performed during Modes 5 and/or 6. The proposed change will eliminate unnecessary mode restrictions on these surveillance requirements.

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

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

Current regulation requires the licensee to responsibly plan, schedule, and perform testing of station equipment. Furthermore, the philosophies of the RSTS [Revised Standard Technical Specifications] do not restrict surveillance performance to specific modes of operation or other plant conditions. Deletion of the mode restrictions will not relinquish licensee responsibility from prudent planning, scheduling, and performance of testing activities and may provide the licensee lower-risk periods of opportunity for test performance. Because of this, the proposed changes are considered to be administrative in nature and do not significantly affect the plant or personnel safety. Modes in which surveillances are performed are not analyzed in association with accident probability or the consequences of an accident. The proposed changes reduce unnecessary restrictions on the licensee and provide consistency with the philosophies of the RSTS.

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

Criterion 2—Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated

The licensee will continue to be accountable for proper and prudent planning, scheduling, and performance of surveillance activities in the absence of the aforementioned mode restrictions proposed for deletion. Therefore, the proposed changes are considered to be administrative in nature and do not significantly affect the plant or personnel safety. The probability of a new or different kind of accident being created remains unchanged since the licensee currently is required to properly plan and execute surveillance tests, even within specific modes of operation. Other activities presently ongoing during the currently specified operational modes could result in an unexpected or unforseen transient or condition if surveillance testing is not properly planned and executed given the other activities in progress and current plant conditions. Since the responsibility of the licensee in these matters remains unchanged by the proposed changes, the possibility of a new or different kind of accident being created also remains unchanged.

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

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

The licensee will continue to be accountable for proper and prudent planning, scheduling, and performance of surveillance activities in the absence of the aforementioned mode restrictions proposed for deletion. Therefore, the proposed changes are considered to be administrative in nature and do not significantly affect the plant or personnel safety. The margin to safety remains unchanged since the licensee currently is required to properly plan and execute surveillance tests, even within specific modes of operation. Other activities presently ongoing during the currently specified operational modes could result in an unexpected or unforseen transient or condition if surveillance testing is not properly planned and executed given the other activities in progress and current plant conditions. Since the responsibility of the licensee in these matters remains unchanged by the proposed changes, no significant reduction in the margin to safety is evident.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: January 21, 2000.

Description of amendment request: Entergy Operations, Inc. requests revision of the Grand Gulf Nuclear Station licensing basis and Technical Specifications to utilize the alternative accident source term described in NUREG–1465.

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:

This proposed amendment to the Grand Gulf Nuclear Station (GGNS) Technical Specifications (TS) revises those specifications affected by the implementation of the alternative source term concepts in accordance with NUREG–1465. In addition, based on the alternative source term, changes are proposed to selected specifications associated with handling irradiated fuel in the primary or secondary containment and CORE ALTERATIONS. Specifically, the proposal uses a new term to describe irradiated fuel that contains sufficient fission products to require operability of accident mitigation systems to meet the accident analysis assumptions. The alternative source term changes affect the definitions and the specifications for the Control Room Fresh Air System, MSIV [main steam isolation valve] leakage surveillance, Standby Gas Treatment System surveillance, and revises a license condition to increase the allowable control room inleakage. The specifications affected by the relaxation of the shutdown controls include those for the Control Room HVAC [heating, ventilation, and air conditioning] system, and the electrical AC [alternating current] Sources, DC [direct current] Sources and Distribution Systems during shutdown.

The Commission has provided standards for determining whether a no significant hazards consideration exists as stated in 10CFR50.92(c). A proposed amendment to an operating license involves a no significant hazards consideration 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.

Entergy Operations, Inc. has evaluated the no significant hazards considerations in its request for a license amendment. In accordance with 10CFR50.91(a), Entergy Operations, Inc. is providing the analysis of the proposed amendment against the three standards in 10CFR50.92(c). A description of the no significant hazards considerations determination follows:

1. The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated.

The alternative source term does not affect the design or operation of the facility; rather, once the occurrence of an accident has been postulated the new source term is an input to evaluate the consequences. The implementation of the alternative source term has been evaluated in revisions to the analyses of the limiting design basis accidents at Grand Gulf Nuclear Station. Based on the results of these analyses, it has been demonstrated that, even with the requested Technical Specification and Operating License changes, the dose consequences of these limiting events are within the regulatory guidance currently proposed by the NRC [Nuclear Regulatory Commission] for use with the alternative source term. This guidance is presented in NUREG–1465, in the draft rulemaking for 10CFR50.67, and in the associated draft Regulatory Guide DG–1081.

A new term to describe irradiated fuel is used to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. These operational conditions are consistent with the design basis analysis. Because the equipment affected by the revised operational conditions is not considered an initiator to any previously analyzed accident, inoperability of the equipment cannot increase the probability of any previously evaluated accident. The proposed requirements bound the conditions of the current design basis fuel handling accident analysis which concludes that the radiological consequences are within the acceptance criteria of NUREG–0800, Section 15.7.4 and General Design Criteria [GDC] 19. As noted above, with the alternative source term implementation, the acceptance criteria are also being revised. The results of the revised Fuel Handling Accident demonstrate that the dose consequences are within the currently proposed NRC regulatory guidance. This guidance is presented in NUREG–1465, in the draft rulemaking for 10CFR50.67, and in the associated draft Regulatory Guide DG–1081.

Therefore, the proposed changes do not significantly increase the probability or consequences of any previously evaluated accident.

2. The proposed changes would not create the possibility of a new or different kind of accident from any previous[ly] analyzed.

The alternative source term does not affect the design, functional performance, or operation of the facility or of any equipment within the facility. Similarly, it does not affect the design or operation of any equipment or systems involved in the mitigation of any accidents. The proposed changes to the Technical Specifications and the Operating License, while they revise certain performance requirements, do not involve any physical modifications to the plant. Therefore, the proposed changes associated with the alternative source term do not create the possibility of a new or different kind of accident from any previous[ly] analyzed.

The new term to describe irradiated fuel is used to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. These operational conditions are consistent with the design basis analyses. The relaxation of selected shut down controls has been modeled in revised analyses. The proposed changes do not introduce any new modes of plant operation and do not involve physical modifications to the plant. Therefore, the proposed changes related to shutdown controls based on the alternative source term do not create the possibility of a new or different kind of accident from any previous[ly] analyzed.

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

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

The changes above are basically associated with the implementation of a new licensing basis for Grand Gulf Nuclear Station. Approval of the basis change from the original source term in accordance with TID–14844 to the new alternative source term of NUREG–1465 is requested by this submittal. The results of the accident analyses revised in support of this submittal, and considering the requested Technical Specification and Operating License changes, are subject to revised acceptance criteria. These analyses have been performed using conservative methodologies as outlined in the currently proposed regulatory guidance. Safety margins and analytical conservatisms have been evaluated and are well understood. The analyzed events have been carefully selected and margin has been retained to ensure that the analyses adequately bound all postulated event scenarios. The dose consequences of these limiting events are within the acceptance criteria also found in the latest regulatory guidance. This guidance is presented in NUREG–1465, in the approved rulemaking for 10CFR50.67, and in the associated draft Regulatory Guide DG–1081.

The proposed changes continue to ensure that the doses at the exclusion area and low population zone boundaries as well as control room, are within the corresponding regulatory limit. In a similar way, the results of the existing analyses demonstrated that the dose consequences were within the applicable NRC-specified regulatory limit. Specifically, the margin of safety for these accidents is considered to be that provided by meeting the applicable regulatory limit, which, for most events, is conservatively set below the 10CFR100 limit. With respect to the control room personnel doses, the margin of safety is the difference between the 10CFR100 limits and the regulatory limit defined by 10CFR50, Appendix A, Criterion 19 (GDC 19).

Therefore, because the proposed changes continue to result in dose consequences within the applicable regulatory limits, they are considered to not result in a significant reduction in a margin of safety.

Based on the above evaluation, operation in accordance with the proposed amendment involves no significant hazards considerations.

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

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: January 24, 2000.

Description of amendment request: Entergy Operations, Inc. requests revisions to the Grand Gulf Nuclear Station Technical Specifications which specify the minimum useable fuel oil inventories to be maintained in the Division 1, 2, and 3 Diesel Generator Fuel Oil Storage Tanks.

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:

Entergy has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR50.92. The following evaluation is provided for the three categories of the significant hazards consideration standards:

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

This change would require additional fuel oil to be stored in each of the Division 1, 2, and 3 Diesel Generator Fuel Oil Storage Tanks. The amount of diesel fuel required to be kept in the storage tanks, which has been determined by Calculation MC–Q1P75–90190 Revision 2 and Calculation MC–Q1P81–90188 Revision 2, is well within the maximum capacity of the Diesel Generator Fuel Oil Storage Tanks. As stated in UFSAR [Updated Final Safety Analysis Report] Section 9.5.4.3 (Safety Evaluation for the diesel fuel storage subsystem) '* * * the tank level will be above the ''seven-day capacity'' required level and will be kept as near the top as practical.'' Other fuel oil storage subsystem components, such as the transfer pumps, are similarly designed, as a minimum, for the storage tanks being filled to maximum capacity. The Diesel Generator Fuel Oil Storage Tanks continue to meet the original design requirements as described in the UFSAR. The proposed change will provide adequate fuel for diesel generator operation at the Technical Specification surveillance testing capacity for Division 1 and 2 Diesel Generators, 5740 KW, and the nameplate rating for Division 3 Diesel Generator, 3300 KW, rather than the lower post-LOCA [loss-of-coolant accident] load profiles previously assumed. Therefore, increasing the quantity of fuel oil required to be maintained, will not increase the probability of the diesel generators becoming an initiator for any previously evaluated accident. Furthermore, since the proposed change increases the fuel oil inventory it should enhance the ability of the diesel generators to respond to an accident and as such the change does not increase the consequences of any previously analyzed accident.

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

The Diesel Generator Fuel Oil subsystem design and operation will not change except for the incorporation of increased fuel oil inventory requirements. This proposed increase remains within the maximum capacity of the Diesel Generator Fuel Oil Storage Tanks. Existing analyses and evaluations, concerning the fuel oil storage tanks, are not adversely impacted by this increase in the required fuel oil inventory. Other fuel oil storage subsystem components, such as the transfer pumps, are similarly designed, as a minimum, for the storage tanks being filled to maximum capacity. The subsystem continues to meet the original design requirements. The proposed increased fuel oil inventory cannot adversely affect any other equipment. Therefore, since the proposed change only increases the fuel oil inventory requirements and does not result in any change in the response of any equipment to an accident, the proposed change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

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

Existing Technical Specification 3.8.3 bases state the Diesel Generator Fuel Oil Storage Tank minimum level is sufficient to operate the respective Diesel Generator for seven days while supplying maximum post-LOCA demands. The proposed change increases the quantity of fuel oil required to be maintained in each of the Division 1, 2, and 3 Diesel Generator Fuel Oil Storage Tanks. The proposed change will provide adequate fuel for diesel generator operation at the Technical Specification surveillance testing capacity for Division 1 and 2 Diesel Generators, 5740 KW, and the nameplate rating for Division 3 Diesel Generator, 3300 KW, rather than the lower post-LOCA load profiles previously assumed. The amount of diesel fuel required to be kept in the storage tanks, which has been determined by Calculation MC–Q1P75–90190 Revision 2 and Calculation MC–Q1P81–90188 Revision 2, is well within the maximum capacity of the Diesel Generator Fuel Oil Storage Tanks. Therefore, since the proposed change increases the fuel oil inventory it should enhance the ability of the diesel generators to respond to an accident and as such the change does not decrease any margin of safety previously assumed.

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

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

Northeast Nuclear Energy Company, et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: January 27, 2000.

Description of amendment request: The proposed amendment would allow operation of the facility for a period of up to 12 hours with the temperature of the ultimate heat sink (UHS) between 75 and 77°F, provided water temperature is verified below 77°F at least once per hour. Currently the temperature limit is 75°F and is verified at least once per 6 hours when the temperature is above 70°F, or once per 24 hours below 70°F.

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:

In accordance with 10 CFR 50.92 NNECO [Northeast Nuclear Energy Company] has reviewed the proposed change and has concluded that it does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed change does not involve a SHC because the change would not:

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

The proposed change will allow plant operation to continue for an additional 12 hours with the temperature of the UHS up to 2°F above the Technical Specification limit of 75°F. This increase in UHS temperature will not affect the normal operation of the plant to the extent which would make any accident more likely to occur. In addition, there exists adequate margin in the safety systems and heat exchangers to assure the safety functions are met at the higher temperature. An evaluation has confirmed that safe shutdown will be achieved and maintained for a loss of coolant accident (LOCA) with a loss of normal power (LNP) and a single active failure with an UHS water temperature as high as 77°F.

The proposed change will have no adverse effect on plant operation, or the availability or operation of any accident mitigation equipment. The plant response to the design basis accidents will not change. In addition, the proposed change can not cause an accident. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change will allow plant operation to continue for an additional 12 hours with the temperature of the UHS up to 2°F above the Technical Specification limit of 75°F. This will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. 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 adverse effect on plant operation or accident mitigation equipment. The proposed change does not introduce any new failure modes. Also, the response of the plant and the operators following these accidents is unaffected by the change. In addition, the UHS is not an accident initiator. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed change will allow plant operation to continue for an additional 12 hours with the temperature of the UHS up to 2°F above the Technical Specification limit of 75°F. An evaluation has been performed which demonstrates that the safety systems have adequate margin to ensure their safety functions can be met with an ultimate heat sink water temperature of 77°F. In addition, safe shutdown capability has been demonstrated for an UHS water temperature as high as 77°F.

The proposed change will have no adverse effect on plant operation or equipment important to safety. The plant response to the design basis accidents will not change and the accident mitigation equipment will continue to function as assumed in the design basis accident analysis. Therefore, there will be no significant reduction in a margin of safety.

The proposed change does not alter the design, function, or operation of the equipment involved. The impact of the proposed change has been analyzed, and it has been determined it does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident from any accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, NNECO has concluded the proposed change does not involve a SHC.

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, Connecticut.

NRC Section Chief: James W. Clifford.

PECO Energy Company, Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: May 26, 1999.

Description of amendment request: The proposed amendments would relocate Technical Specification (TS) Surveillance Requirement 4.1.3.5.b, regarding the performance of channel functional test and channel calibration of certain control rod scram accumulator instrumentation, to the Updated Final Safety Analysis Report and would revise TS 3.1.3.5 to allow an alternate method for verifying whether a control rod drive pump is operating.

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

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

The first proposed change relocates control rod drive (CRD) instrumentation requirements from the TS to the UFSAR and plant procedures. The second proposed change adds an alternate method for verifying operation of a control rod drive pump in the TS action statement.

Regarding the first proposed change, operability of the accumulators is determined by verifying that the pressure in each accumulator is greater than or equal to 955 psig. TS 4.1.3.5.a requires weekly verification of accumulator pressure. The local pressure indicator for each accumulator is the normal means of satisfying this surveillance. This proposed change does not affect or alter the requirements associated with this instrumentation. If the local pressure indicator is not functioning or pressure is less than 955 psig, the accumulator will still be declared inoperable.

Operability of the accumulator pressure or water level alarm and indication function provided by the Reactor Manual Control System (RMCS) is not critical to the ability to insert control rods because:

(1) The rods can be inserted with normal charging water pressure if the accumulator is inoperable;

(2) A controlled shutdown or scram would occur before the accumulator would lose its full capability to insert the control rod, if it is found that no control rod drive pumps are operating according to existing procedural and TS controls placed on the plant; and

(3) The subject instruments' alarm and indication function are part of routine operational monitoring and are not considered in the plant safety analysis.

[Therefore, the removal of the accumulator pressure or level indication does not impact the consequences or probability of an accident previously evaluated. The operational monitoring of the accumulator alarms and indication system affords operating personnel the status of system condition and the opportunity to initiate appropriate actions if deemed necessary.]

The second proposed change simply adds an alternate method for verifying operation of a control rod drive pump. This check provides an equivalent method of verifying that inoperable control rod accumulators were not caused by a control rod drive pump trip. In addition:

(1) The assumed control rod reactivity insertion rate is not changed;

(2) The maximum number of inoperable accumulators and control rods is not changed;

(3) The TS actions to be taken in the event that a control rod drive pump is not operating remain unchanged; and

(4) The instrumentation for accumulator leakage a pressure detection will continue to be maintained and calibrated.

A RMCS failure does not change the failure modes or the reliability of the control rod function as described and evaluated in the UFSAR. The CRD system will continue to be available to safely shutdown the plant as described and evaluated in the UFSAR.

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

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

Neither the mechanism for initiating nor for carrying out a scram is modified by either of these proposed changes. These proposed changes do not:

(1) Create a means by which the scram function could be impeded or prevented.

(2) Involve a physical plant alteration or change the methods governing normal plant operation.

(3) Impose or eliminate any requirements or change the controls for maintaining the requirements.

There are no other malfunctions that need to be considered since failure of a significant number of control rods to scram is analyzed in Section 15.8 of the UFSAR. This is the bounding analysis for multiple control rod malfunctions or severe degradation of control rod scram performance. This event is mitigated by safety systems not directly related to the CRD system including the scram accumulators.

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

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

The first proposed change relocates CRD instrumentation requirements from TS to the UFSAR and plant procedures. The proposed change will not reduce a margin of safety, because it has no impact on any safety analysis. * * * [Therefore, the proposed change does not involve a significant reduction in a margin of safety.]

The second proposed change adds an alternate method for verifying operation of a control rod drive pump in the TS action statement. This proposed change does not reduce a margin of safety because the proposed change does not:

(1) Affect the maximum allowable control rod scram times,

(2) Change the maximum allowable number or minimum separation of inoperable control rods, or

(3) Modify any of the instrument setpoints or functions.

The proposed change will either maintain the present margin of safety or increase it, by reducing the need for unnecessary challenges to the Reactor Protection System (RPS) and resulting plant shutdown, while still maintaining the capability to complete a reactor scram.

Therefore, these proposed TS 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.

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, PA 19464.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: James W. Clifford.

Power Authority of the State of New York, Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: February 3, 2000.

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) by revising the reactor water level setpoint for the Anticipated Transient Without Scram Recirculation Pump Trip (ATWS–RPT) function and the Alternate Rod Injection (ARI) functions (Table 3.2–7).

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:

Operation of the FitzPatrick plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

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

The proposed TS change deals only with an instrumentation setpoint which initiates the ATWS–RPT/ARI function. The system is intended to provide a mitigation function during a postulated ATWS event and does not provide any other plant control function. However, if the ATWS–RPT/ARI system were to fail, the result would be a trip of the recirculation pumps, or reactor scram, both of which are currently evaluated. The design of the system includes a one-out-of-two-twice logic, which ensures that a single failure in the system cannot cause or inhibit the ATWS–RPT/ARI function. Therefore, the probability of an inadvertent recirculation pump trip or inadvertent reactor scram is not changed from the event as currently described in the JAFNPP UFSAR [James A. FitzPatrick Nuclear Power Plant Updated Final Safety Analysis Report].

FitzPatrick specific analyses were performed by General Electric Company with NRC approved methods for postulated ATWS events (Reference 1 [''James A. FitzPatrick Nuclear Power Plant Anticipated Transient Without Scram Analysis, for Recirculation Pump Trip Setpoint Changes,'' General Electric Company, NEDC–32616P, July 18, 1996, Previously Docketed with NRC]). The specific events evaluated include the Main Steamline Isolation Valve closure event, Inadvertent Opening of a Relief Valve, and the Loss of Feedwater. For these events, the following acceptance criteria were established:

Peak Reactor Pressure (maximum 1 SRV out of service)—< 1500 psig

Peak Suppression Pool Temperature—< 190°F

Fuel Remains Cooled—Coolant Level > TAF [Top of Active Fuel]

The analyses demonstrate that all criteria were adequately met with the proposed TS change implemented, further ensuring no increase in the consequences of the postulated events.

The basis for changing the ARI initiation setpoint on reactor level to be consistent with that proposed for the ATWS RPT is documented in Reference 2 [JAF–ICD–NBI–;03998, Rev. 0—Alternate Rod Insertion Setpoint (an internal FitzPatrick interface document)]. The ARI initiation point is not specified in the Technical Specification.

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

The proposed TS change deals only with a reactor water level instrumentation setpoint, which initiates the ATWS–RPT/ARI function. The existing level transmitters and wiring will be used, and new analog trip units will be incorporated which are identical to existing low-low reactor water level trip units currently shared with HPCI [High Pressure Coolant Injection] and RCIC [Reactor Core Isolation Cooling] initiation. These new analog trip units are of a different design (General Electric) than those used in the Reactor Protection System (Rosemount) and therefore, the diversity requirement of 10 CFR 50.62 (c)(3) remain[s] satisfied. This allows the HPCI and RCIC setpoints to remain the same while only lowering the ATWS–RPT/ARI setpoint. The sensing, logic and actuation of the ATWS–RPT/ARI design is not modified. This includes the use of the existing one-out-of-two-twice logic, which ensures that a single failure in the circuit will not cause or inhibit the ATWS–RPT/ARI function. There are no new signals required as input, and the trip function is accomplished with the existing RPT breakers and existing scram pilot air header solenoid valves. The system does not provide input to any other plant function. The plant will not operate in any new mode nor are there any new operational requirements as a result of the proposed change. Therefore, it is not considered possible for the ATWS–RPT/ARI system to fail in any new or different way from those events currently evaluated in the JAFNPP UFSAR.

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

The ATWS–RPT/ARI function protects the fuel, reactor and containment from failure during a postulated ATWS event. The fuel cladding barrier is protected via adequate cooling, provided by ensuring that the core remains covered throughout the entire event. The reactor coolant system boundary is protected by ensuring compliance with the ASME [American Society of Mechanical Engineers] emergency class pressure limit of 120% of design pressure. The containment is protected by ensuring the suppression pool temperature limits are met.

FitzPatrick specific ATWS analyses were performed by postulating events that challenge each of these limits (Reference 1). With the proposed TS change considered, each of these limits were met without a need for any reduction in the margin of safety established in the JAFNPP UFSAR for the primary fission product barriers.

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

Attorney for licensee: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Section Chief: Marsha Gamberoni, Acting.

Public Service Electric & Gas Company, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: February 24, 2000.

Description of amendment request: The proposed amendment would approve a revision to the Hope Creek Generating Station Updated Final Safety Analysis Report (UFSAR) to reflect the use of the Mechanical Vacuum Pumps (MVPs) to evacuate the condenser during plant startup at power levels less than or equal to 5%. These revisions are required to make the UFSAR accident analyses associated with a Control Rod Drop Accident (CRDA) consistent with actual plant operation. Public Service Electric and Gas Company (PSE&G) has performed an engineering calculation that demonstrates that there is an increase in the radiological consequences of a CRDA coincident with MVP operation. Nuclear Regulatory Commission (NRC) approval of the proposed UFSAR changes is required, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.59, since these changes involve an unreviewed safety question.

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

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

The Condenser Air Removal System has no safety-related function and its failure does not jeopardize the function of any safety-related system or component or prevent a safe shutdown of the plant. Neither the MVPs, nor other components associated with the Condenser Air Removal, Gaseous Radwaste Off-Gas, Process Radiation Monitoring, or Turbine Building HVAC [Heating, Ventilation, and Air Conditioning] systems or the South Plant Vent are design basis accident initiators. The operation of mechanical vacuum pump at power levels ≤ 5% will not increase the probability of occurrence of a main condenser air removal system leak or failure of the line leading to the steam jet air ejector (SJAE) near the main condenser. Additionally, the design and operation of the condenser off-gas system is not impacted. Moreover, MVP operation will not increase the probability of occurrence of a CRDA or any other design basis accident. Consequently, this proposal does not increase the probability of an accident previously evaluated.

The engineering calculation performed to assess the impact of the use of the MVPs demonstrated that the radiological consequences of a CRDA coincident with MVP operation increase but remain well within the 10CFR100 guidelines and meet SRP [Standard Review Plan] Section 15.4.9, Appendix A, acceptance criteria. Additionally, the calculation demonstrated that the radiological consequences of a CRDA coincident with MVP operation are within the GDC [General Design Criterion] 19 guidelines for control room personnel and plant operators and remain bounded by the loss of coolant accident analysis for on-site personnel. Therefore, although the proposal does increase the consequences of a CRDA, the proposal does not significantly increase the consequences of an accident previously evaluated.

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

The proposal involves crediting manual action to trip the MVPs; however, PSE&G has evaluated this operator action against the criteria in NRC Information Notice 97–78 and has concluded that adequate controls are in place to ensure that the subject manual action is taken. In addition, the proposal does not change monitor setpoints, affect equipment qualification, or otherwise create an accident initiator not previously considered. Consequently, this proposal does not create the possibility of an accident of a different type from any previously evaluated.

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

The Condenser Air Removal System has no safety-related function. Failure of the system does not jeopardize the function of any safety-related system or component or prevent a safe shutdown of the plant.

The radiological activity evaluated in this proposal does not result in scenarios that could impact 10 CFR 50 Appendix I, 10 CFR 20, or 40 CFR 190 release criteria. Post-scram shutdown or startup condition MVP operation in accordance with plant operating procedures will not degrade the original design for the Condenser Air Removal System.

An engineering calculation was prepared that demonstrated that the radiological consequences of a CRDA coincident with MVP operation remain well within the 10 CFR 100 guidelines and that the consequences meet SRP Section 15.4.9, Appendix A, acceptance criteria. Additionally, the engineering calculation demonstrated that the radiological consequences of a CRDA coincident with MVP operation are within GDC 19 guidelines for control room personnel and plant operators and remain bounded by the loss of coolant accident analysis for on-site personnel.

Since no design bases are degraded, the Technical Specifications operating limits, that provide sufficient operating range such that the acceptance limits are not exceeded during plant operations and analyzed transients, are not [ ] affected. Since the acceptance limits are not exceeded, implementation of this proposal does not reduce the margin of safety as described in the basis for any Technical Specifications.

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

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038. *NRC Section Chief:* James W. Clifford.

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

Date of amendment requests: February 23, 2000 (PCN 508).

Description of amendment requests: The amendment application is a request to allow an option regarding the methodology for measuring the reactivity worth of control element assembly (CEA) groups for San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 during low-power physics testing following a refueling. The proposed option involves measuring the worth of approximately three-fourths of the full-length CEA groups each refueling cycle rather than the present methodology, which measures the worth of all full-length CEA groups each refueling cycle. Measured CEA groups would be rotated such that each full-length group would be measured at least every other refueling. The licensee has determined this change to involve an unreviewed safety question.

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

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

No. The proposed option to the Low Power Physics Test (LPPT) program will involve performance of rod worth measurements of typically six of eight full-length control element assembly (CEA) groups each refueling, rather than performance of rod worth measurements of all eight CEA groups each refueling. Thus, the LPPT option will result in a reduction in the number of plant manipulations required for LPPT. Inverse Boron Worth (IBW) is not required in the proposed LPPT program option, but it may be determined during the performance of a boration or dilution, which is already a part of the present LPPT program. The manipulations which will be performed are a subset of the evolutions which are performed in the existing test sequence. Therefore, the LPPT testing option does not carry any increased risk of any accident evaluated in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR). Since the number and duration of manipulations are reduced, there would actually be a small reduction in accident potential.

The proposed test program option will not compromise the technical objectives of the LPPT program. Fuel fabrication, core and reactor internals reassembly, CEA worths, mechanical integrity and reliability, performance of core physics design codes and consistency with design and safety analysis expectations will remain validated with the same effectiveness as is achieved in the current program. In addition, the reduced duration of operation in the LPPT Special Test Exception of the Technical Specifications has a positive impact on nuclear safety.

Therefore, the proposed LPPT program option does not involve a significant increase in the probability of an accident previously evaluated.

The proposed test program option will eliminate CEA exchange measurements and determine CEA worth by dilution/boration measurements. Measurement of CEA worth by the dilution/boration methods achieves typically higher quality results than the CEA Exchange method.

The proposed LPPT program option does not include the requirement to measure inverse boron worth. However, a measured initial critical boron concentration and measured CEA group worths that match predicted values within acceptance criteria are sufficient to verify adequate core physics modeling without a separate IBW measurement.

Since the proposed test sequence option continues to ensure that core operation and reactivity control are consistent with design expectations, the proposed LPPT option will not involve a significant increase in the consequences of an accident previously evaluated.

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

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

No. The proposed LPPT program option does not create any plant condition or manipulation which is materially different from those of the existing program. Furthermore, the number of manipulations and duration of Special Test Exceptions are significantly reduced. The proposed LPPT program option relies entirely on conventional boration and dilution rod worth measurement test methods which have been industry standards. The methodology used to measure IBW, if performed, does not introduce any new evolutions during LPPT and cannot create a new or different type of accident.

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

(3) Does this amendment request involve a significant reduction in a margin of safety?

No. The proposed LPPT program option fully achieves objectives of the reload test program by validating fuel fabrication, core reassembly, CEA worths, mechanical integrity and reliability, performance of physics design codes and consistency with design and safety analysis expectations with the same effectiveness as is achieved in the current program. As a result, all assumptions made in support of UFSAR Chapter 15 Safety Analyses regarding CEA performance remain valid.

The effectiveness of the SONGS 2 & 3 Reload Test program, including LPPT and Power Ascension Testing, has been evaluated and shown to be uncompromised by the proposed LPPT option. Specific testing requirements imposed by the Nuclear Regulatory Commission are captured in Technical Specification Surveillance Requirements. The proposed LPPT program option is fully compliant with existing Technical Specification Surveillance Requirements and validates the core physics models regarding core performance, reactivity control and proper core reassembly to an extent equivalent to that of the present program.

The proposed LPPT program option is also consistent with the recently modified ANSI/ANS 19.6.1–1997 standard for Pressurized Water Reactor reload testing, with the exception of the requirement and methodology to determine IBW. The ANSI/ANS standard was developed with participation from industry and NRC representatives and represents an expert panel assessment of what is appropriate for an LPPT program. A measured initial critical boron concentration and measured CEA group worths that match predicted values within acceptance criteria are sufficient to verify adequate core physics modeling, and infer that the IBW value is within standard acceptance critieria, without a separate IBW measurement.

Therefore, the proposed LPPT program option 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 requests involve no significant hazards consideration.

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

NRC Section Chief: Stephen Dembek.

Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: August 24, 1999, as supplemented on December 29, 1999.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) 3.3.2 ''Engineered Safety Feature Actuation System (ESFAS) Instrumentation'' to relax the slave relay test frequency from quarterly to a refueling frequency.

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

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

The results of WCAP–13878 demonstrate that slave relays are highly reliable. WCAP–13878 also provides guidance to assure that slave relays remain highly reliable. The aging assessment concludes that the age/temperature-related degradation of all ND relays, and NE relays produced after 1992, is sufficiently slow such that a refueling frequency surveillance interval will not significantly increase the probability of slave relay failures. Finally, the evaluation of the auxiliary relays actuated during slave relay testing has concluded that based on the tests of the auxiliary relays performed during other equipment testing, reasonable assurance is provided that failures will be identified if the associated slave relays are tested on a refueling frequency.

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

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

The proposed changes do not alter the performance of the ESFAS mitigation systems assumed in the plant safety analysis. Changing the interval for periodically verifying ESFAS slave relays (assuring equipment operability) will not create any new accident initiators or scenarios.

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

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

The proposed changes do not affect the total ESFAS response assumed in the safety analysis since the reliability of the slave relays will not be significantly affected by the decreased surveillance frequency.

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: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Section Chief: Richard L. Emch, Jr.

STP Nuclear Operating Company, Docket No. 50–499, South Texas Project, Unit 2, Matagorda County, Texas

Date of amendment request: February 21, 2000.

Description of amendment request: STP Nuclear Operating Company proposes to amend the South Texas Project (STP), Unit 2 technical specifications (TS) so that steam generator tube eddy-current inspection indications of less than or equal to 3.0 volts can be left in service if found at intersections of tube hot-leg tube-support-plates C through M (3.0-volt alternate repair criteria). The new alternate repair criteria would apply only until the Unit 2 Model E steam generators are replaced during the outage currently scheduled to commence in fall of 2002. STP Nuclear Operating Company also proposes to amend the STP, Unit 2 TS to make an editorial correction to Note 1 and Note 2, on page 3/4–16a to align the notes with the preceding paragraph. STP Nuclear Operating Company also provided, for information only, changes to the Bases for TS 3/4.4.5 to provide the structural margins and Westinghouse topical report references used as the bases for the use of the 3.0-volt alternate repair criteria.

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:

In accordance with the criteria set forth in 10 CFR 50.92, the STP Nuclear Operating Company (STPNOC) has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. Conformance of the proposed amendment to the standards for a determination of no significant hazard as defined by the criteria set forth in 10 CFR 50.92 is shown in the following discussions addressed to each criterion:

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

During the limiting design-basis steam-line-break (SLB) event, South Texas Project (STP) Unit 2 steam generator tube burst criteria are inherently satisfied for marginally degraded (primarily axially-oriented ODSCC [outer diameter stress corrosion cracking]) tube spans at certain tube support plate (TSP) intersections.

Steam generator tubes pass through holes drilled in the TSP. The inside diameter (ID) of the drilled holes closely approximates the outside diameter (OD) of the tubes. Generally, the TSP precludes those tube spans within the drilled holes from deforming beyond the diameters of the drilled holes, thus, precluding tube burst in the restrained regions. However, design basis SLB events may vertically displace a TSP, removing its support from the tube spans passing through it. For TSP C through M, maximum displacement during a postulated SLB event is less than 0.15 inch. Because TSP C through M remain essentially stationary during all conditions, tube spans included within the drilled holes are restrained during the limiting SLB event. Thus, the tube burst margin for intersections of tube hot-legs and TSP C through M is independent of voltage related growth rates and the proposed 3-volt ARC [alternate repair criteria] is compliant with RG [Regulatory Guide] 1.121 [Bases for Plugging Degraded PWR Steam Generator Tubes] criteria.

Given a TSP displacement of < 0.15 inch, tube hot-leg spans enclosed within TSP C through M have a negligible tube burst probability of less than 10^{-10} for a single tube. This is eight orders of magnitude less than the 10^{-2} probability-of-burst criterion specified by GL [Generic Letter] 95–05 [Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking] and represents negligible axial tube burst probabilities for tube hot-leg spans intersecting TSP C through M. Thus, repair limits to preclude burst are not needed and tube repair limits may be based primarily on limiting leakage to acceptable levels during accident conditions.

Cracks that include cellular corrosion may yield to axial loads, resulting in tensile tearing of the tube at that location. A tensile load requirement to prevent this establishes a structural limit for the tube expansion based plugging criterion. In order to establish a lower bound for the structural limit, tensile tests were used to measure the force required to separate a tube that exhibits cellular corrosion. Additionally, pulled tubes with cellular and/or inter-granular attack (IGA) tube wall degradation were evaluated and the tensile strength of the tube conservatively calculated from the remaining non-corroded cross-section of the tube. This calculation assumes that the degraded portions contribute nothing to the axial load carrying ability of the tube. Data from these tests shows that circumferential cracks exhibiting bobbin-coil-probe-indication-voltages greater than 35 volts require tube-pressure-differentials well above the operating limit of 3-times-normal differential pressure in order to produce circumferential ruptures (i.e., axial separation at the plane of the crack). This proposal specifies a structural limit of 17 volts (safety factor of 2) to ensure conservative results for repairs at intersections of tubes with TSP C through M.

GL 95–05 states that licensees must perform SLB leak rate and tube burst probability analyses before returning to power from outages during which they perform steam generator inspections. Licensees must include the results in a report to the NRC within 90 days after restart. If an analysis reveals that leak-rate or burst-probability exceeds limits, the licensee must report it to the NRC and assess the safety significance of this finding. Model E steam generator SLB leak rates are calculated for indications found at intersections of tube hot-legs and TSP. Both SLB leak rate and tube burst probability are calculated for tube hot-leg intersections with FDB [flow distribution baffles], hot-leg intersections with TSP N through R, and indications found at intersections of tube cold-legs with any TSP.

It has been established that the design basis main SLB outside of containment and upstream of the MSIV [main steam isolation valves] produces the limiting radiological consequence from any tube leakage that may be postulated to exist at the initiation of an accident. With use of 3-volt ARC, STPNOC [STP Nuclear Operating Company] will calculate the maximum primary-to-secondary leakage for the last day of the coming steam generator service-cycle and use this value to calculate the radiological consequence of the limiting SLB event. This methodology will ensure that site boundary doses for this accident remain within an acceptable fraction of the 10 CFR 100 guidelines and that doses to the control room operators remain within GDC 19 [10 CFR Part 50, Appendix A, General Design Criterion] limits.

Based on the above, STPNOC concludes that operation of South Texas Project Unit 2 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Use of the proposed steam generator tube 3-volt ARC does not significantly change circumstances or conclusions assumed by the plant design basis. Application of the 3-volt ARC does not significantly increase the probability of either single or multiple tube ruptures. Steam generator tube integrity remains adequate for all plant operating conditions.

STPNOC has confirmed that the allowed post-accident primary-to-secondary leakage rate for SLB events results in the limiting offsite and control room doses for South Texas Project Unit 2. A projected SLB leak rate of 15.4 gpm is calculated to produce doses 90% of the currently licensed South Texas Project Unit 2 dose limits (Reference 2 [STPNOC letter dated July 15, 1998, NOC–AE–000228, Response to NRC Request for Additional Information related to STP Unit 2 Amendment No. 83]). STPNOC TS impose a normal leak rate limit of 150 gpd (0.1 gpm) per steam generator to minimize the potential for excessive leakage during all plant conditions. The 150 gpd limit provides added margin to accommodate contingent leakage should a stress corrosion crack grow at a greater than expected rate or extend outside the TSP. Leakage trending consistent with EPRI Report TR–04788, ''PWR Primary-to-Secondary Leak Guidelines'' has been established for South Texas Project Unit 2.

Since steam generator tube integrity will meet GL 95–05 requirements and be confirmed through in-service inspection and primary-to-secondary leakage monitoring, the proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

RG 1.121 describes a method for meeting GDC 14, 15, 31, and 32 by reducing the probability or consequences of steam-generator tube-rupture through application of criteria for removing degraded tubes from service. These criteria set limits of degradation for steam generator tubing through in-service inspection. Analyses show that tube integrity will remain consistent with the criteria of Regulatory Guide 1.121 after implementation of the proposed 3-volt ARC. Even under the worst case ODSCC occurrence at TSP elevations, 3-volt ARC will not cause or significantly increase [the] probability of a steam-generator tube-rupture event.

In addressing combined LOCA [loss-of-coolant accident] + SSE [safe-shutdown earthquake] effects on steam generator components as required by GDC 2, analysis has shown that tube collapse may occur in certain regions of the steam generators of some plants. This collapse is caused by TSP plastic deformation in the region of the TSP wedge supports. Plastic deformation occurs when TSP experience large lateral loads concentrated at wedge support points on the periphery of a TSP undergoing combined loading effects of a LOCA rarefaction wave and SSE. Deformation impinges on TSP apertures through which tubes pass, deflecting tube walls inward. The resulting pressure differential across deformed tube walls may cause some tubes to collapse.

There are two issues associated with steam generator tube collapse. First, collapse of steam generator tubing reduces RCS [reactor coolant system] flow. RCS flow reduction increases resistance to heat flow from the core during a LOCA, increasing Peak Clad Temperature (PCT). Second, partial through-wall tube-cracks could become full through-wall tube-cracks during tube deformation or collapse. Tubes in regions affected by this phenomenon are usually excluded from evaluation under 3-volt ARC. STP Model E steam generator design does not produce this plastic deformation, thus is not subject to tube collapse. No STP Unit 2 tubes are excluded, for this reason, from application of the proposed 3-volt ARC.

End of Cycle (EOC) distribution of crack indications at affected TSP elevations will be confirmed to allow no more than the acceptable primary-to-secondary leakage rate during all plant conditions and not adversely affect radiological dose consequences. For the limiting SLB event, STPNOC will calculate leak rates as free-span leakage for ODSCC indications at tube and TSP intersections. The calculations will use GL 95–05 leak rate methods with an additional component for potentially overpressurized indications [discussed in detail in the Safety Evaluation section of the licensee's February 21, 2000, application under the heading ''SLB Leak Rate and Tube Burst Probability Considerations''].

Inspections conducted in accordance with RG 1.83, Rev. 1 [In Service Inspection of Pressurized Water Reactor Steam Generator Tubes], using 3-volt ARC for intersections of tube hot-legs with TSP C through M, and using 1-volt ARC at remaining hot-leg and cold-leg intersections will be supplemented by:

(1) enhanced eddy current inspection procedures to achieve consistency in voltage normalization,

(2) eddy current inspection of 100% of tubes found, using inspection of a 20% tube sample, to have ODSCC at intersections with TSP, and

(3) a required RPC [rotating pancake coil] inspection of the larger indications to confirm that the principal degradation mechanism continues to be ODSCC.

Plugging steam generator tubes reduces RCS flow margin. As previously noted, increasing repair limits for indications found at TSP intersections will reduce the number of tubes that must be plugged. Thus, 3-volt ARC will conserve RCS flow margin, preserving operational and safety benefits that would otherwise be reduced by unnecessary plugging.

Therefore, the proposed license amendment does not result in a significant increase in dose consequences represented in the current licensing basis, and does not involve a significant reduction in margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the standards of 10 CFR 50.92(c) are satisfied. The staff also reviewed the proposed editorial change for no significant hazards consideration. The proposed editorial correction does not affect the design or operation of the facility and satisfies the three standards of 10 CFR 50.92(c). Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

Attorney for licensee: Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Robert A. Gramm.

Virginia Electric and Power Company, Docket Nos. 50–280 and 50–281, Surry Power Station, Units No. 1 and No. 2, Surry County, Virginia

Date of amendment request: November 29, 1999.

Description of amendment request: The proposed changes will modify the Technical Specifications (TS) in Section 3.23 for the Main Control Room and Emergency Switchgear Room Ventilation and Air Conditioning Systems; TS Surveillance Requirement Sections 4.20, Basis 4.20.A.7, and 4.20.B.4 for the Control Room Air Filtration System; and TS Surveillance Requirement Sections 4.12.A.6, 4.12.A.7, 4.12.A.8, 4.12.B.7, and 4.12 Basis for the Auxiliary Ventilation Exhaust Filter Trains. The proposed changes will revise the above Surveillance Requirements for the laboratory testing of the carbon samples for methyl iodide removal efficiency to be consistent with American Society for Testing and Materials (ASTM) Standard D3803–1989, ''Standard Test Method for Nuclear-Graded Activated Carbon,'' with qualification, as the laboratory testing standard for both new and used charcoal adsorbent used in the ventilation system.

Basis for proposed no significant hazards consideration determination: In 10 CFR 50.92, three criteria are provided to determine whether a proposed license amendment involves a significant hazards consideration. No significant hazards consideration is involved if operation of the facility 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. 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:

Virginia Electric and Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed changes for Surry Units 1 and 2 and determined that a significant hazards consideration is not involved. The proposed Technical Specification changes adopt the nuclear-grade charcoal testing requirements of ASTM D3803–1989, with qualification, for methyl iodide removal efficiency and the requirements of ASTM D3803–1979, with qualification, for elemental iodine removal efficiency. The method of testing nuclear-grade activated charcoal does not affect the design or operation of the plant. The changes also do not involve any physical modification to the plant or result in a change in a method of system operation. The adoption of the 1989 edition of ASTM D3803 for methyl iodide testing conforms with approved guidance for testing of nuclear-grade activated charcoal. This provides assurance that testing of ventilation systems is being performed with a suitable standard to ensure that charcoal adsorbers are capable of performing their required safety function and that the regulatory requirements regarding onsite and offsite dose consequences continue to be satisfied. The changes do not create an unreviewed safety question.

(a) The proposed changes modify surveillance testing requirements and do not affect plant systems or operation and therefore do not increase the probability or the consequences of an accident previously evaluated. The proposed surveillance requirements adopt ASTM D3803–1989, with qualification, as the laboratory method for testing samples of the charcoal adsorber for methyl iodide removal efficiency in response to NRC's Generic Letter 99–02. This method of testing charcoal adsorbers has been approved by the NRC as an acceptable method for determining methyl iodide removal efficiency. Since the charcoal adsorbers are used to mitigate the consequences of an accident, the more accurate the test, the better assurance we have that we remain within our accident analysis assumptions. Testing of the charcoal adsorbers' efficiency for removing elemental iodine is performed in accordance with the 1979 version of ASTM D3803 since the 1989 version does not address elemental iodine removal efficiencies. The laboratory test acceptance criteria contain a safety factor to ensure that the efficiency assumed in the accident analysis is still valid at the end of the operating cycle. There is no change in the method of plant operation or system design.

(b) The proposed changes modify surveillance testing requirements and do not impact plant systems or operations and therefore do not create the possibility of an accident or malfunction of a different type than evaluated previously. The proposed surveillance requirements adopt ASTM D3803–1989, with qualification, as the laboratory method for testing samples of the charcoal adsorber for methyl iodide removal efficiency. This change is in response to NRC's request in Generic Letter 99–02. Testing of the charcoal adsorbers' efficiency for removing elemental iodine is performed in accordance with the 1979 version of ASTM D3803 since the 1989 version does not address elemental iodine removal efficiencies. There is no change in the method of plant operation or system design. There are no new or different accident scenarios, transient precursors, nor failure mechanisms that will be introduced.

(c) The proposed changes modify surveillance test requirements and do not impact plant systems or operations and therefore do not significantly reduce the margin of safety. The revised surveillance requirements adopt ASTM D3803–1989, with qualification, as the laboratory method for testing samples of the charcoal adsorber for methyl iodide removal efficiency. The 1989 edition of this standard imposes very stringent requirements for establishing the capability of new and used activated carbon to remove methyl iodide from air and gas streams. The results of this test provide a more conservative estimate of the performance of nuclear-graded activated carbon used in nuclear power plant HVAC [heating, ventilation, and air conditioning] systems for the removal of methyl iodide. Testing of the charcoal adsorbers' efficiency for removing elemental iodine is performed in accordance with the 1979 version of ASTM D3803 since the 1989 version does not address elemental iodine removal efficiencies. The laboratory test acceptance criteria contain a safety factor to ensure that the efficiency assumed in the accident analysis is still valid at the end of the operating cycle.

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. Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Section Chief: Richard L. Emch, Jr.

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.

Consumers Energy Company, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan

Date of amendment request: February 18, 2000.

Brief description of amendment request: The amendment changes current Technical Specification (TS) 4.9a.2 and improved TS 3.7.5 and its associated bases to remove requirements associated with the backup steam supply to turbine-driven auxiliary feedwater pump P–8B.

*Date of publication of individual notice in **Federal Register:*** March 1, 2000 (65 FR 11089)

Expiration date of individual notice: Comment period expired March 14, 2000; Notice period expires March 31, 2000.

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

Date of application for amendments: February 25, 2000.

Brief description of amendments: The amendment revises Technical Specification Table 3.3.2–1, ''Engineered Safety Feature Actuation System Instrumentation'' to provide a one-time exception, until the next time the turbine is removed from service, from the requirement to perform response time testing for the solenoid valve 1–FSV–47–027.

*Date of publication of individual notice in the **Federal Register:*** March 2, 2000.

Expiration date of individual notice: March 16, 2000.

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

Date of application for amendment: January 14, 2000, as supplemented by letter dated February 17, 2000 (ULNRC–04172 and –04187).

Brief description of amendment request: The amendment would revise several sections of the improved Technical Specification (ITSs) to correct 14 editorial errors made in either (1) the application dated May 15, 1997, (and supplementary letters) for the ITSs, or (2) the certified copy of the ITSs that was submitted in the licensee's letters of May 27 and 28, 1999. The ITSs were issued as Amendment No. 133 by the staff in its letter of May 28, 1999, and will be implemented by the licensee to replace the current TSs by April 30, 2000.

*Date of publication of individual notice in **Federal Register:*** February 25, 2000 (65 FR 10118).

Expiration date of individual notice: March 27, 2000.

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, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of application for amendment: March 1, 1999.

Brief description of amendment: The amendment approves changes to the Updated Safety Analysis Report concerning design requirements for physical protection from tornado missiles.

Date of issuance: February 29, 2000.

Effective date: February 29, 2000.

Amendment No.: 124.

Facility Operating License No. NPF–62: The amendment allows a change to the Updated Safety Analyis Report concerning tornado missile protection.

*Date of initial notice in **Federal Register:*** April 21, 1999 (64 FR 19558).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 29, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 4, 1999, as supplemented December 13, 1999.

Brief description of amendment: The amendment modified the limiting conditions for operation in the Technical Specifications (TSs) under which a reduction in the number of means of decay heat removal (DHR) capability may occur by deleting two of these conditions. The amendment also makes related Bases changes and clarifies the DHR requirements for redundancy.

Date of issuance: February 28, 2000.

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

Amendment No.: 220.

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

*Date of initial notice in **Federal Register:*** June 30, 1999 (64 FR 35207). The December 13, 1999, letter withdrew a Bases change of the June 4, 1999, application and did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 28, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 26, 1999.

Brief description of amendment: The amendment authorized changes to Chapters 5 and 14 of the Updated Final Safety Analysis Report (UFSAR). The changes reflect the use of an Electric Power Research Institute-developed Conservative Deterministic Failure Margin methodology for seismic analysis of the portions of the nonsafety-related auxiliary steam line piping located in the Auxiliary, Control, and Fuel Handling buildings at TMI–1.

Date of issuance: March 10, 2000.

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

Amendment No.: 221.

Facility Operating License No. DPR–50. Amendment authorizes changes to the UFSAR.

Date of initial notice in ***Federal Register:*** June 30, 1999 (64 FR 35207). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 10, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 8, 1999, as supplemented July 20 and November 24, 1999.

Brief description of amendments: The amendments revise the Technical Specifications to increase the storage capacity of spent fuel in the fuel storage pools by allowing credit for soluble boron and decay time in the safety analysis, and to increase the maximum radially averaged fuel enrichment from 4.3 weight percent to 4.8 weight percent.

Date of issuance: March 2, 2000.

Effective date: March 2, 2000.

Amendment Nos.: Unit 1–125, Unit 2–125, Unit 3–125.

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

Date of initial notice in ***Federal Register:*** September 20, 1999 (64 FR 50835). The July 20 and November 24, 1999, letters provided additional clarifying information that was within the scope of the original application and **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 2, 2000.

No significant hazards consideration comments received: No.

Carolina Power & Light Company, et al., Docket No. 325, Brunswick Steam Electric Plant, Unit 1, Brunswick County, North Carolina

Date of amendment request: September 28, 1999.

Brief description of amendment: The amendment changes the Technical Specifications (TS) in response to your submittal dated September 28, 1999. The amendment revises TS 2.1.1.2, ''Reactor Core Safety Limits,'' and TS 5.6.5, ''Core Operating Limits Report,'' by removing safety limit restrictions which are no longer applicable.

Date of issuance: March 1, 2000.

Effective date: March 1, 2000.

Amendment No.: 207.

Facility Operating License No. DPR–71: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 3, 1999 (64 FR 59797).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 2000.

No significant hazards consideration comments received: No.

Carolina Power & Light Company, et al., Docket No. 50–325, Brunswick Steam Electric Plant, Unit 1, Brunswick County, North Carolina

Date of amendment request: November 17, 1999.

Brief description of amendment: The amendment changes the Technical Specifications (TS) in response to the licensee's submittal dated September 28, 1999. The amendment revises TS 2.1.1.2, ''Reactor Core Safety Limits,'' by changing the Minimum Critical Power Ratio.

Date of issuance: March 1, 2000.

Effective date: March 1, 2000.

Amendment No.: 208.

Facility Operating License No. DPR–71: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 15, 1999 (64 FR 70080).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 2000.

No significant hazards consideration comments received: No.

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois; Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: March 23, 1999, as supplemented on October 21, 1999, and December 15, 1999.

Brief description of amendments: The amendments approved the installation of new Boral high density spent fuel storage racks at Byron and Braidwood stations. The amendments also approved an increase in the spent fuel pool storage capacity from 2,870 assemblies to 2,984 assemblies at each station.

Date of issuance: March 1, 2000.

Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 112 and 105.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** June 16, 1999 (64 FR 32280). The October 21 and December 15, 1999, supplements did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 1, 2000.

No significant hazards consideration comments received: No.

Commonwealth Edison Company, 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 12, 1999.

Brief description of amendments: The amendments revised Technical Specification (TS) 2.2, ''Limiting Safety System Settings,'' and TS 3/4.1.A, ''Reactor Protection System,'' to remove an anticipatory reactor scram signal, the turbine electro-hydraulic control (EHC) low oil pressure trip, from the reactor protection system trip function requirements.

Date of issuance: January 28, 2000.

Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 193 & 189.

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

Date of initial notice in ***Federal Register:*** December 1, 1999 (64 FR 67331).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 28, 2000.

No significant hazards consideration comments received: No.

Commonwealth Edison Company, 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: November 16, 1999.

Brief description of amendments: The amendments change Technical Specification Table 4.1.A–1, ''Reactor Protection System Instrumentation Surveillance Requirements,'' to modify the surveillance requirements for Functional Unit 3, ''Reactor Vessel Steam Dome Pressure—High,'' to reflect replacement of the pressure switches with analog trip units.

Date of issuance: January 28, 2000.

Effective date: Immediately, to be implemented before startup from Refueling Outage 16 for Unit 1 and before startup from Refueling Outage 15 for Unit 2.

Amendment Nos.: 194 & 190.

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

Date of initial notice in ***Federal Register:*** December 15, 1999 (64 FR 70082).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 28, 2000.

No significant hazards consideration comments received: No.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: June 2, 1999, as supplemented August 25, 1999.

Brief description of amendment: The amendment allows for the relocation of the Quality Assurance related administrative controls to the Quality Assurance Program Description in accordance with NRC Administrative Letter 95–06, ''Relocation of Technical Specification Administrative Controls Related to Quality Assurance.''

Date of issuance: February 25, 2000.

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

Amendment No.: 206.

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

Date of initial notice in ***Federal Register:*** November 3, 1999 (64 FR 59799).

The August 25, 1999, letter provided clarifying information that did not change the initial proposed no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 25, 2000.

No significant hazards consideration comments received: No.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: February 29, 2000.

Brief description of amendment: The amendment revises Technical Specification (TS) 3.7.D.1 to correct an editorial error, TS 6.2.2 to change the senior reactor operator license requirement for the Operations Manager, and TS 6.3.1 to modify the qualification requirement for the Operations Manager.

Date of issuance: February 29, 2000.

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

Amendment No.: 207.

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

Date of initial notice in ***Federal Register:*** April 7, 1999 (64 FR 17023).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 29, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 24, 1999, as supplemented by letter dated November 24, 1999.

Brief description of amendments: The amendments revised the Technical Specifications by revising the minimum reactor coolant system (RCS) flow rate limit, the reactor coolant average temperature, and the pressurizer pressure limits, and by restricting operation to a RCS flow deficit of no more than one percent.

Date of issuance: March 1, 2000.

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

Amendment Nos.: Unit 1—184; Unit 2—176.

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

Date of initial notice in ***Federal Register:*** August 11, 1999 (64 FR 43770).

The November 24,1999, letter provided clarifying information that did not change the scope of the June 24, 1999, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 1, 2000.

No significant hazards consideration comments received: No

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

Date of application for amendments: June 24, 1999, as supplemented by letter dated November 24, 1999.

Brief description of amendments: The amendments revise the minimum reactor coolant system (RCS) flow rate limit, reduce the reactor coolant average temperature and pressurizer pressure limits, restrict operation to a RCS flow deficit of no more than one percent, and change the low RCS flow reactor trip setpoint.

Date of issuance: March 2, 2000.

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

Amendment Nos.: Unit 1—191; Unit 2—172.

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

Date of initial notice in ***Federal Register:*** August 11, 1999 (64 FR 43772).

The November 24, 1999, supplemental letter did not expand the scope of the application initially noticed or change the proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 2, 2000.

No significant hazards consideration comments received: No.

Energy Northwest, Docket No. 50–397, WNP–2, Benton County, Washington

Date of application for amendment: October 13, 1999.

Brief description of amendment: The amendment removes footnote (d) from Function 5, ''RHR [residual heat removal] SDC [shut down cooling] System Isolation'' of Technical Specification (TS) Table 3.3.6.1–1, ''Primary Containment Isolation Instrumentation.'' Footnote (d) states, ''Only the inboard trip system is required in Modes 1, 2, and 3, as applicable, when the outboard valve control is transferred to the alternate remote shutdown panel and the outboard valve is closed.'' The outboard suction trip system valve, RHR–V–8, is no longer transferred to the alternate remote shutdown panel and is now required during Modes 1, 2 and 3. Therefore, footnote (d) is no longer needed. Footnote (e) is relettered as footnote (d) for consistency.

Date of issuance: March 9, 2000.

Effective date: March 9, 2000, to be implemented within 30 days of issuance.

Amendment No.: 161.

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

Date of initial notice in ***Federal Register:*** December 15, 1999 (64 FR 70082).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 9, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 16, 1999.

Brief description of amendment: The amendment authorizes the licensee to revise fuel handling accident (FHA) dose calculations for three scenarios described in the River Bend Station, Unit 1, Updated Safety Analysis Report. The first is an FHA in the fuel building, assumed to occur 24 hours post-shutdown. A second FHA analysis was prepared to support Amendment 35 to RBS Technical Specifications (TS) which assumed an FHA occurs in the primary containment 80 hours post-shutdown during local leakage rate testing (LLRT). A third analysis was prepared in support of Amendment 85 to the River Bend Station Technical Specifications which assumed the containment is open at 11 days. These analyses are being updated to account for several changes that were determined by the licensee to involve an unreviewed safety question in accordance with Title 10 of the *Code of Federal Regulations,* Section 50.59(a)(2)(i).

Date of issuance: March 2, 2000.

Effective date: The license amendment is effective as of its date of issuance and shall be implemented in the next periodic update to the USAR in accordance with 10 CFR 50.71(e). Implementation of the amendment is the incorporation into the USAR update, the changes to the description of the facility as described in the licensee's application dated December 16, 1999, and evaluated in the staff's Safety Evaluation attached to this amendment.

Amendment No.: 110.

Facility Operating License No. NPF–47: The amendment authorized changes to the Updated Safety Analysis Report.

Date of initial notice in ***Federal Register:*** January 26, 2000 (65 FR 4272).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 2, 2000.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket Nos. 50–313 and 50–368, Arkansas Nuclear One, Units 1 and 2, Pope County, Arkansas

Date of amendment request: September 17, 1999.

Brief description of amendments: The amendments modify TS 3.25.2, ''Radioactive Gas Storage Tanks,'' at Arkansas Nuclear One, Unit 1 (ANO–1) and TS 3/4.11.2, ''Gas Storage Tanks,'' at Arkansas Nuclear One, Unit 2 (ANO–2). This change will reduce the limiting condition for operation for the maximum quantity of stored radioactivity per tank from 300,000 curies of noble gases as Xenon–133 (Xe–133) equivalent to 78,782 curies of noble gases as Xe–133 equivalent at ANO–1, and 82,400 curies of noble gases as Xe–133 equivalent at ANO–2.

Date of issuance: February 18, 2000.

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

Amendment Nos.: ANO–1—204; ANO–2—211.

Facility Operating License Nos. DPR–51 and NPF–6: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 12, 2000 (65 FR 1921).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 18, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: July 29, 1998, as supplemented by letters dated July 29, October 28, and November 11, 1999

Brief description of amendment: The amendment replaces the existing reference to the Asea Brown Boveri-Combustion Engineering, Inc. small break loss-of-coolant accident emergency core cooling system performance evaluation model with the revised model described in the topical report CENPD–137, Supplement 2, P–A, April 1998.

Date of issuance: March 7, 2000.

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

Amendment No.: 158.

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

Date of initial notice in ***Federal Register:*** December 15, 1999 (64 FR 70085).

The July 29, October 28, and November 11, 1999, letters provided additional information that did not change the scope of the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 7, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 16, 1999.

Brief description of amendments: This amendment revised TS 3/4.7.1.3 and associated Bases for the Primary Plant Demineralized Water (PPDW) system to clarify that the minimum specified volume of water in the PPDW Storage Tank is a usable volume. Additionally, the minimum usable volume of water in the PPDW Storage Tank is increased, and a clarifying footnote that the specified value is an analysis value is added. Finally, several editorial and administrative changes, such as revision of action statement wording, addition of license number to TS page, and addition of clarifying information to the TS Bases regarding analysis assumptions are made.

Date of issuance: February 28, 2000.

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

Amendment Nos.: 106.

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

Date of initial notice in ***Federal Register:*** April 21, 1999, (64 FR 19556).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 28, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: May 27, 1999.

Brief description of amendments: The amendments relocate the seismic monitoring instrumentation requirements contained in Technical Specification (TS) 3/4.3.3.3 to the Licensing Requirements Manual (LRM) based on the guidance provided in Generic Letter 95–10, ''Relocation of Selected Technical Specifications Requirements Related to Instrumentation.'' The Bases section for Specification 3/4.3.3.3 is also relocated to the LRM. The appropriate Index pages, Table Index page (Unit No. 1 only), TS pages and Bases pages are revised to reflect the removal of the seismic monitoring instrumentation specification from the TSs. An additional TS page is added to reflect that TS Number 3/4.3.3.4 is not used.

This additional page also denotes the number of the following page. Finally, the Bases section is modified to denote that TS Number 3/4.3.3.4 is not used.

Date of issuance: February 28, 2000.

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

Amendment Nos.: 228 and 107.

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

*Date of initial notice in **Federal Register:*** June 30, 1999 (64 FR 35203).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 28, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: May 27, 1999.

Brief description of amendments: The amendments (1) revised the frequency for performing the CHANNEL FUNCTIONAL TEST of the manual initiation functional units specified in the Beaver Valley Power Station, Unit Nos. 1 and 2, Engineered Safety Features Actuation System (ESFAS) Instrumentation Technical Specifications (TSs) from monthly, with an accompanying footnote which allows the manual initiation to be tested on a refueling interval, to each refueling interval; (2) revise footnotes associated with TS ESFAS tables; (3) revise associated TS Bases.

Date of issuance: February 28, 2000.

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

Amendment Nos.: 229 and 108.

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

*Date of initial notice in **Federal Register:*** June 30, 1999 (64 FR 35205).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 28, 2000.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of application for amendment: May 21, 1999, as supplemented by submittals dated December 1, 1999, and January 28, 2000.

Brief description of amendment: This amendment revises the Technical Specifications to expand the present spent fuel storage capability by 289 storage locations by allowing the use of spent fuel racks in the cask pit area adjacent to the spent fuel pool.

Date of issuance: February 29, 2000.

Effective date: February 29, 2000.

Amendment No.: 237.

Facility Operating License No. NPF–3: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 8, 1999 (64 FR 36933).

The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 29, 2000.

No significant hazards consideration comments received: No.

Florida Power and Light Company, et al., Docket No. 50–389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of application for amendment: August 18, 1999.

Brief description of amendment: This amendment decreases the surveillance frequency, listed in the updated Final Safety Analysis Report (UFSAR), for cycling steam valves in the turbine overspeed protection system from monthly to quarterly.

Date of Issuance: February 28, 2000.

Effective Date: As of the date of its issuance, to be incorporated into the UFSAR at the time of its next update.

Amendment No.: 108.

Facility Operating License No. NPF–16: Amendment revised the UFSAR.

*Date of initial notice in **Federal Register:*** September 22, 1999 (64 FR 51345).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 28, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 1, 1999, as supplemented December 15, 1999.

Breif description of amendments: The amendments revised License Condition 3.L for Turkey Point, Units 3 and 4, Operating Licenses DPR–31 and DPR–41 to reflect the December 1, 1999, date of the last revision to the Physical Security Plan. Also, the phrase ''Turkey Point Plant, Units 3 and 4 Security Plan'' was revised to ''Turkey Point Physical Security Plan.''

Date of issuance: February 28, 2000.

Effective date: February 28, 2000.

Amendment Nos.: 204 and 198.

Facility Operating License Nos. DPR–31 and DPR–41: Amendments revised the Operating Licenses.

*Date of initial notice in **Federal Register:*** December 29, 1999 (64 FR 73092).

The Commssion's related evaluation of the amendments is contained in a Safety Evaluation dated February 28, 2000.

No significant hazards consideration comments received: No.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of application for amendment: October 12, 1999.

Brief description of amendment: The amendment revises the Technical Specifications, Appendix B, ''Environmental Protection Plan (Non-Radiological)'' to incorporate the reasonable and prudent measures, and the terms and conditions, of the Incidental Take Statement in the Biological Opinion issued by the National Marine Fisheries Service.

Date of issuance: February 29, 2000.

Effective date: February 29, 2000.

Amendment No.: 190.

Facility Operating License No. DPR–31: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 15, 1999 (64 FR 70090).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 29, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 22, 1999.

Brief description of amendments: The amendments delete Technical Specification 5.4.2, ''Reactor Coolant System Volume,'' regarding the reactor coolant system (RCS) volume information. Information concerning the RCS volume is included in the D. C. Cook Updated Final Safety Analyses Report (UFSAR), and any changes to the information are controlled in accordance with 10 CFR 50.59.

Date of issuance: March 1, 2000.

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

Amendment Nos.: 241 and 222.

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

*Date of initial notice in **Federal Register:*** January 13, 2000 (65 FR 2199).

The Commssion's related evaluation of the amendments is contained in a Safety Evaluation dated March 1, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: October 6, 1999, as supplemented February 9, 2000.

Brief description of amendment: The amendment addresses the following changes to the Technical Specifications: (1) provisions for implementation of 10 CFR Part 50, Appendix J, Option B, (Technical Specification Task Force (TSTF) Change 52, Revision 2) (2) extension of the required surveillance interval for the containment air lock interlock mechanism from 18 to 24 months (TSTF Change 17, Revision 1), (3) clarification of the valve types requiring isolation time testing (TSTF Change 46, Revision 1), and (4) provisions for use of administrative means for verification of isolation devices that are locked, sealed or otherwise secured (TSTF Change 269, Revision 2).

Date of issuance: March 3, 2000.

Effective date: March 3, 2000, to be implemented within 30 days.

Amendment No.: 180.

Facility Operating License No. DPR–46: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 29, 1999 (64 FR 73092). The February 9, 2000, supplement provided clarifying information that was within the scope of the October 6, 1999, application and the staff's original **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commssion's related evaluation of the amendment is contained in a Safety Evaluation dated March 3, 2000.

No significant hazards consideration comments received: No.

Niagara Mohawk Power Corporation, Docket No. 50–220, Nine Mile Point Nuclear Station, Unit 1, Oswego County, New York

Date of application for amendment: August 26, 1999, as supplemented December 17, 1999.

Brief description of amendment: The amendment changes Technical Specification 3.2.3, ''Coolant Chemistry,'' to support the implementation of noble metal chemical addition.

Date of issuance: March 8, 2000.

Effective date: As of the date of issuance to be implemented before the licensee first performs the noble metal chemical addition.

Amendment No.: 169.

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

*Date of initial notice in **Federal Register:*** September 22, 1999 (64 FR 51347).

The licensee's supplemental letter dated December 17, 1999, did not change the Commission's finding of no significant hazards consideration.

The Commssion's related evaluation of the amendment is contained in a Safety Evaluation dated March 8, 2000.

No significant hazards consideration comments received: No.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: October 25, 1999, as supplemented on February 2 and 7, 2000.

Brief description of amendment: The amended Technical Specifications permit use of the already-installed Oscillation Power Range Monitor system.

Date of issuance: March 2, 2000.

Effective date: As of the date of issuance to be implemented before activation of the Oscillation Power Range Monitor System, but no later than August 31, 2000.

Amendment No.: 92.

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

*Date of initial notice in **Federal Register:*** December 1, 1999 (64 FR 67336).

The February 2 and 7, 2000, letters provided clarifying information that did not change the initial proposed no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 2, 2000.

No significant hazards consideration comments received: No.

Northeast Nuclear Energy Company, et al., Docket No. 50–245, Millstone Nuclear Power Station, Unit No. 1, New London County, Connecticut

Date of application for amendments: April 19, 1999, as supplemented August 25, October 14, November 3, December 20, 1999, and February 29, 2000.

Brief description of amendments: The amendment replaces the current Technical Specifications for fuel storage pool water lever, crane operability, and crane travel with a spent fuel cask with new Technical Specifications to reflect the permanently defueled status of the plant.

Date of Issuance: March 7, 2000.

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

Amendment No.: 107.

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

*Date of initial notice in **Federal Register:*** June 30, 1999, (64 FR 35208).

The August 25, October 14, November 3, December 20, 1999, and February 29, 2000, letters provided clarifying information that did not change the scope of the original application and proposed no hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 7, 2000.

No significant hazards consideration comments received: No.

Northeast Nuclear Energy Company, et al., Docket Nos. 50–336 and 50–423, Millstone Nuclear Power Station, Unit Nos. 2 and 3, New London County, Connecticut

Date of application for amendment: November 23, 1999.

Brief description of amendment: The amendment changes Technical Specification (TS) 4.0.5, ''Limiting Conditions for Operation and Surveillance Requirements'' by adding a biennial or 2-year surveillance interval and incorporating a required frequency for performing inservice testing activities of once per 731 days.

Date of issuance: March 8, 2000.

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

Amendment Nos.: 241 and 178.

Facility Operating License Nos. DPR–65 and NPF–49: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 26, 2000 (65 FR 4286).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 8, 2000.

No significant hazards consideration comments received: No.

Northern States Power Company, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota

Date of application for amendments: March 2, 1998, supplemented on January 21, 2000.

Brief description of amendments: The amendments change the second paragraph of Technical Specification 3.8.D, ''Spent Fuel Pool Special Ventilation System,'' to clarify restrictions on movement of loads in the spent fuel pool enclosure with one train of spent fuel pool special ventilation system inoperable.

Date of issuance: February 17, 2000.

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

Amendment Nos.: 147 and 138.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** May 20, 1998 (63 FR 27763).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 17, 2000.

No significant hazards consideration comments received: No.

Northern States Power Company, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota

Date of application for amendments: November 6, 1996, supplemented April 10 and October 1, 1997, and March 4, 1998.

Brief description of amendments: The amendments revise Technical Specification Section 5.0, ''DESIGN FEATURES,'' by relocating certain portions of the design features information to the Updated Safety Analysis Report, consistent with NUREG–1431, ''Standard Technical Specifications, Westinghouse Plants,'' Revision 1.

Date of issuance: February 29, 2000.

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

Amendment Nos.: 148 and 139.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 29, 1997 (62 FR 4338).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 29, 2000.

No significant hazards consideration comments received: No.

Public Service Electric & Gas Company, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: August 26, 1999.

Brief description of amendment: This amendment raises the condensate storage tank (CST) low level setpoint and the corresponding allowable value in Technical Specification Tables 3.3.3–2 and 3.3.5–2. The subject setpoint is associated with the automatic transfer of the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) pump suctions from the CST to the suppression pool in the event of low CST level. These changes are being made to address concerns regarding potential vortexing in the HPCI and RCIC suction flowpaths.

Date of issuance: March 6, 2000.

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

Amendment No.: 124.

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

Date of initial notice in ***Federal Register:*** September 22, 1999 (64 FR 51348).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 6, 2000.

No significant hazards consideration comments received: No.

Public Service Electric & Gas Company, 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: July 29, 1999, as supplemented November 30, 1999.

Brief description of amendments: The amendments revise Technical Specifications Surveillance Requirement 4.6.1.1 to clarify when verification of primary containment integrity may be performed by administrative means and to change the surveillance interval for verification of manual valves and blind flanges inside of containment.

Date of issuance: February 29, 2000.

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

Amendment Nos.: 227 and 208.

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

Date of initial notice in ***Federal Register:*** September 22, 1999 (64 FR 51349).

The November 30, 1999, letter provided clarifying information that did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 29, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 11, 1996 (PCN 460), as supplemented April 6, 1998, and March 22 and July 29, 1999.

Brief description of amendments: The amendments revise Technical Specification 3.6.3, ''Containment Isolation Valves,'' to specify that the completion time for required action for certain containment isolation valves be in accordance with the applicable limiting condition for operation pertaining to the engineered safety features system in which they are installed.

Date of issuance: March 9, 2000.

Effective date: March 9, 2000, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2–165; Unit 3–156.

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

Date of initial notice in ***Federal Register:*** January 19, 2000 (65 FR 2993), as corrected January 26, 2000 (65 FR 4265).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 9, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 13, 1999, as supplemented February 24, 2000 (PCN–507).

Brief description of amendments: The amendments revise the license expiration dates for San Onofre Unit 2 to February 16, 2022, and for San Onofre Unit 3 to November 15, 2022, thus extending the units' periods of operation to the full 40-year design-basis lifetime.

Date of issuance: March 9, 2000.

Effective date: March 9, 2000, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2—166; Unit 3—157.

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

Date of initial notice in ***Federal Register:*** December 29, 1999 (64 FR 73098).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 9, 2000.

No significant hazards consideration comments received: No

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

Date of amendment request: September 30, 1998, as supplemented May 14 and October 21, 1999.

Brief description of amendments: The amendments revise the South Texas Project, Units 1 and 2, offsite dose licensing bases to account for (1) operation of the existing steam generators at reduced feedwater inlet temperatures and (2) operation with the new replacement steam generators, also at a reduced feedwater temperature. The changes revised calculated offsite doses for four existing Updated Final Safety Analysis Report (UFSAR) Chapter 15 accidents and added a discussion in Chapter 15 of the radiological analysis for the voltage-based criteria for steam generator tubes.

Date of issuance: March 2, 2000.

Effective date: March 2, 2000, to be implemented within 30 days.

Amendment Nos.: Unit 1—124; Unit 2—112

Facility Operating License Nos. NPF–76 and NPF–80: Amendments authorize revisions to the UFSAR.

Date of initial notice in ***Federal Register:*** November 18, 1998 (63 FR 64124).

The May 14 and October 21, 1999, supplemental letters provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 2, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 30, 1999, as supplemented January 13, 2000.

Brief description of amendments: The amendments revise the Technical Specifications (TS) to delete the necessity for time response testing various instrument transmitters based on historical records indicating satisfactory time responses in the past.

Date of issuance: February 29, 2000.

Effective date: February 29, 2000.

Amendment Nos.: Unit 1—251; Unit 2—242.

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

Date of initial notice in ***Federal Register:*** October 6, 1999 (64 FR 54381). The supplemental letter of January 13, 2000, did not expand the scope of the initial amendment request or change the NRC staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 29, 2000.

No significant hazards consideration comments received: No

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

Date of application for amendments: October 14, 1999 as supplemented February 23 and March 2, 2000.

Brief description of amendments: Revise Section 4.4 of the Technical Specification (TS) surveillance testing requirements and their associated Bases to incorporate an alternate repair criteria for axial primary water stress corrosion cracking at dented tube support plate intersections.

Date of issuance: March 8, 2000.

Effective date: March 8, 2000.

Amendment Nos.: Unit 1—252; Unit 2—243.

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

Date of initial notice in ***Federal Register:*** December 29, 1999 (64 FR 73100). The supplemental letters dated February 23, and March 2, 2000, did not expand the scope of the original amendment request or change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 8, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 25, 1999, as supplemented December 17, 1999.

Brief description of amendment: The amendment revises the main steam safety valve Technical Specification (TS) Section 3.7.1 to provide a new requirement to reduce the power range neutron flux-high reactor trip setpoints when two or more main steam safety valves (MSSVs) per steam generator are inoperable.

Date of issuance: March 7, 2000.

Effective date: March 7, 2000.

Amendment No.: 19.

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

Date of initial notice in ***Federal Register:*** August 11, 1999 (64 FR 43781). The letter dated December 17, 1999 provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 7, 2000.

No significant hazards consideration comments received: No

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

Date of amendment request: February 11, 1999, as supplemented by letters dated September 3 and December 20, 1999.

Brief description of amendments: The amendments change the Technical Specifications to authorize an increase in the allowable spent fuel storage capacity and the crediting of soluble boron, in the spent fuel pool, for spent fuel reactivity control.

Date of issuance: February 24, 2000.

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

Amendment Nos.: 74.

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

Date of initial notice in ***Federal Register:*** May 12, 1999 (64 FR 25522).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 24, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 20, 2000.

Brief description of amendment: The amendment redefines the functional testing criteria for the noble gas activity monitor instrumentation in the Augmented Off-Gas system.

Date of Issuance: March 6, 2000.

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

Amendment No.: 184.

Facility Operating License No. DPR–28: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 2, 2000 (65 FR 4999).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated March 6, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 11, 2000.

Brief description of amendment: The amendment deletes the requirement to exercise the main steam isolation valves (MSIVs) twice weekly by partial closure and subsequent re-opening. Testing of the MSIVs to demonstrate their safety function will continue to be performed on a quarterly basis in accordance with the Vermont Yankee Inservice Testing program, Technical Specifications (TSs), and applicable provisions of Section XI of the ASME Boiler and Pressure Vessel Code. The TS change is issued as a follow-up amendment to NOED 00–06–01, which was orally granted on February 10, 2000.

Date of Issuance: March 9, 2000

Effective date: As of the date of issuance, and shall be implemented prior to March 25, 2000.

Amendment No.: 185

Facility Operating License No. DPR–28: Amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards considerations: Yes (65 FR 8749) February 22, 2000. That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by March 23, 2000, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 9, 2000.

No significant hazards consideration comments received: No

Virginia Electric and Power Company, et al., Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of application for amendments: May 6, 1999, as supplemented June 22 and December 16, 1999.

Brief description of amendments: The amendments revise the Technical Specifications Sections 3.3.1.1; 4.3.1.1.1; 4.3.1.1.2; 4.3.1.1.3; 3.3.2.1; 4.3.2.1.1; 4.3.2.1.2; 4.3.2.1.3; 3/4.3.1; 3/4.3.2 and 6.8.4.9 and Tables 3.3–1; 4.3–1; 3.3–3 and 4.3–2 for Unit 1, and Sections 3.3.1.1; 4.3.1.1.1; 4.3.1.1.2; 4.3.1.1.3; 3.3.2.1; 4.3.2.1.1; 4.3.2.1.2; 4.3.2.1.3; 3/4.3.1; 3/4.3.2 and 6.8.4.9 and Tables 3.3–1; 4.3–1; 3.3–3 and 4.3–2 for Unit 2, to revise the surveillance frequency for the Reactor Trip System (RTS) and the Engineered Safety Features Actuation System (ESFAS) analog instrumentation channels. In addition, the allowed outage time and action times for the RTS and ESFAS analog instrumentation and the actuation logic are being modified.

Date of issuance: March 9, 2000

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

Amendment Nos.: 221 and 202.

Facility Operating License Nos. NPF–4 and NPF–7. Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** June 16, 1999 (64 FR 32291). The letters of June 22 and December 16, 1999, contained clarifying information only, and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 9, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 15, 1999.

Brief description of amendment: The amendment modified the improved technical specifications (ITS) that were issued in Amendment No. 123 on March 31, 1999, and implemented on December 18, 1999. The changes expand the region of acceptable reactor coolant pump (RCP) seal injection flow to each RCP in Figure 3.5.5–1 and provides 10 editorial changes to the ITS.

Date of issuance: March 1, 2000.

Effective date: March 1, 2000, to be implemented within 60 days of the date of issuance.

Amendment No.: 132.

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

Date of initial notice in ***Federal Register:*** January 26, 2000 (65 FR 4292).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 2000.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 15th day of March 2000.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–6913 Filed 3–21–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Draft Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission has issued for public comment a draft of a new guide in its Regulatory Guide Series. This series has been developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits and licenses.

The draft guide, temporarily identified by its task number, DG–1075 (which should be mentioned in all correspondence concerning this draft guide), is titled ''Emergency Planning and Preparedness for Nuclear Power Reactors.'' This guide is being developed to propose guidance on methods acceptable to the NRC staff for complying with the NRC's regulations for emergency response plans and preparedness at nuclear power reactors.

This draft guide has not received complete staff approval and does not represent an official NRC staff position.

Comments may be accompanied by relevant information or supporting data. Written comments may be submitted to the Rules and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW., Washington, DC. Comments will be most helpful if received by May 22, 2000.

You may also provide comments via the NRC's interactive rulemaking