NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to section 189a. (2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. The Act requires the Commission publish notice of any amendments issued, or proposed to be issued and 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 October 29, 2004, through November 12, 2004. The last biweekly notice was published on November 9, 2004 (69 FR 64984).

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. Within 60 days after the date of publication of this notice, 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.

Normally, the Commission will not issue the amendment until the expiration of 60 days after the date of publication of this notice. The Commission may issue the license amendment before expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. In addition, the Commission may issue the amendment prior to the expiration of the 30-day comment period should circumstances change during the 30-day comment period such that failure to act in a timely way would result, for example in derating or shutdown of the facility. Should the Commission take action prior to the expiration of either the comment period or the notice period, it will publish in the **Federal Register** a notice of issuance. Should the Commission make a final No Significant Hazards Consideration Determination, any hearing will take place after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. (Note: Public access to ADAMS has been temporarily suspended so that security reviews of publicly available documents may be performed and potentially sensitive information removed. Please check the NRC Web site for updates on the resumption of ADAMS access.) The filing of requests for a hearing and petitions for leave to intervene is discussed below.

Within 60 days after the date of publication of this notice, 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 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. (**Note:** Public access to ADAMS has been temporarily suspended so that security reviews of publicly available documents may be performed and potentially sensitive information removed. Please check the NRC Web site for updates on the resumption of ADAMS access.) If a request for a hearing or petition for leave to intervene is filed within 60 days, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.309, 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 general requirements: (1) The name, address and telephone number of the requestor or petitioner; (2) the nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; (3) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and (4) the possible effect of any decision or order which may be entered in the proceeding on the requestor's/petitioner's interest. The petition must also set forth the specific contentions which the petitioner/requestor seeks to have litigated at the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner/requestor shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner/requestor intends to rely in proving the contention at the hearing. The petitioner/requestor must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner/requestor intends to rely to establish those facts or expert opinion. The petition must include 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/requestor to relief. A petitioner/requestor who fails to satisfy 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.

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, 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 by: (1) First class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff; (2) courier, express mail, and expedited delivery services: Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff; (3) E-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, *HEARINGDOCKET@NRC.GOV;* or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC, Attention: Rulemakings and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

Nontimely requests and/or petitions and contentions will not be entertained absent a determination by the Commission or the presiding officer of the Atomic Safety and Licensing Board that the petition, request and/or the contentions should be granted based on a balancing of the factors specified in 10 CFR 2.309(a)(1)(i)–(viii).

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

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

Date of amendment request: October 20, 2004.

Description of amendment request: The proposed change would revise Technical Specification (TS) Table 4.1–1 functional testing surveillance interval from monthly to semi-annually for the following reactor protection system instrument channels: Table 4.1–1, Item No. 4, ''Power Range Channel,'' Item No. 7, ''Reactor Coolant Temperature Channel,'' Item No. 8, ''High Reactor Coolant Pressure Channel,'' Item No. 9, ''Low Reactor Coolant Pressure Channel,'' Item No. 10,'' Flux-Reactor Coolant Flow Comparator,'' Item No. 11, ''Reactor Coolant Pressure-Temperature Comparator,'' Item No. 12, ''Pump Flux Comparator,'' Item No. 13, ''High Reactor Building Pressure Channel,'' Item No. 45, ''Loss of Feedwater Reactor Trip,'' and Item No. 46, ''Turbine Trip/Reactor Trip.'' The TS Section 4.1 Bases would be revised to reflect the proposed change from monthly to semi-annually and to specify that one channel is being tested every 46 days on a continual sequential rotation, which is consistent with the calculations of BAW–10167A, Supplement 1, and associated Nuclear Regulatory Commission Safety Evaluation Report that indicate that the reactor protection system retains a high level of reliability for this test interval. The proposed change would also revise TS Table 4.1–1 functional testing surveillance interval from monthly to quarterly for the following reactor protection system reactor trip devices: Table 4.1–1, Item No. 1, ''Protection Channel Coincidence Logic,'' and Item No. 2, ''Control Rod Drive Trip Breaker and Regulating Rod Power SCRs.'' The TS Section 4.1 Bases would be revised to reflect the proposed change from monthly to quarterly testing and to specify that one channel is being tested every 23 days on a continual sequential rotation, which is consistent with the calculations of BAW–10167A, Supplement 3, February 1998, and the NRC SER for BAW–10167A, Supplement 3, dated January 7, 1998, that indicate that the reactor trip system retains a high level of reliability for this test interval.

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

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

Response: No.

The reactor protection system monitors parameters related to safe operation and trips the reactor to protect the reactor core against fuel cladding damage. It also assists in protecting against reactor coolant system damage caused by high system pressure by limiting energy input to the system through reactor trip action. Therefore, this change has no impact on the probability of an accident previously evaluated. The results of the reliability analyses conducted in accordance with NRC [Nuclear Regulatory Commission] approved methodology and criteria show that the test interval extension of the reactor protection system instrument channels and reactor trip devices is not a significant contributor to trip system unavailability or the risk of core damage. The reactor protection system instrument channel and reactor trip device functional test surveillance program will continue to ensure that the reactor protection system is capable of performing its intended safety function during a design basis accident.

Therefore, this change has no effect on the consequences of an accident previously evaluated.

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

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

Response: No.

The proposed change involves the reactor protection system instrument channel and reactor trip device surveillance test interval, which is not, in and of itself, considered to be an accident initiator. Postulated failure of the reactor protection system instrument channel or reactor trip device to function is an analyzed condition and does not constitute a new or different kind of accident. The proposed change does not create any new failure modes not bounded by previously analyzed accidents.

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

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

Response: No.

The results of the reliability analysis conducted in accordance with NRC approved methodology and criteria show that the test interval extension of the reactor protection system instrument channels and reactor trip devices is not a significant contributor to trip system unavailability or the risk of core damage. The Technical Specifications will continue to require the reactor protection system trip setpoints to remain within the assumptions of the accident analysis and that adequate reliability of the reactor protection system trip devices is maintained, thus preserving existing margins of safety.

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

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

Attorney for licensee: Thomas S. O'Neill, Associate General Counsel, AmerGen Energy Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: August 20, 2004.

Description of amendment request: The proposed amendment would revise the Allowable Values for the following Reactor Protection System (RPS) instrumentation functions: Intermediate Range Neutron Flux, Reactor Coolant Flow—Low, Steam Generator Water Level—Low Coincident with Steam Flow/Feedwater Flow Mismatch, and Intermediate Range Neutron Flux (P–6) Interlock. Additionally, these changes revise the Allowable Value for the Engineered Safety Feature Actuation System Instrumentation function for High Steam Flow in Two Steam Lines Coincident with Steam Line Pressure—Low. Also the proposed amendment would delete an unnecessary footnote associated with the applicability for the Automatic Trip Logic RPS instrumentation function.

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 proposal to revise the Allowable Values for the affected reactor protection and engineered safety feature actuation functions was developed in accordance with the current setpoint methodology for HBRSEP [H. B. Robinson Steam Electric Plant], Unit No. 2, thus ensuring that the probability and consequences of previously evaluated accidents are not significantly increased. The proposed deletion of the unnecessary footnote associated with the Automatic Trip Logic reactor protection instrumentation function does not change the requirements for operability of this function. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated, because the factors that are used to determine the probability and consequences of accidents are not being affected.

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

The proposed changes will continue to ensure that the operability of the previously described functions will be appropriately maintained. No physical changes to the HBRSEP, Unit No. 2, systems, structures, or components are being implemented. There are no new or different accident initiators or sequences being created by the proposed Technical Specifications changes. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes, as previously described, ensure that the margin of safety for the applicable fission product barriers that are protected by these functions will continue to be maintained. This conclusion is based on the use of a valid setpoint methodology for determining the Allowable Values for the reactor protection and engineered safety feature actuation functions. Therefore, these changes do not involve a significant reduction in the margin of safety.

Based on the preceding discussion, the requested changes do not involve a significant hazards consideration.

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

Attorney for licensee: Steven R. Carr, Associate General Counsel—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Michael L. Marshall.

Duke Energy Corporation, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina; Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1and 2, Mecklenburg County, North Carolina; Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of amendment request: September 28, 2004.

Description of amendment request: The proposed amendment would delete Technical Specification (TS) 5.6.1, ''Occupational Radiation Exposure Report,'' and TS 5.6.4, ''Monthly Operating Reports.''

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on June 23, 2004 (69 FR 35067). The licensee affirmed the applicability of the model NSHC determination in its application dated September 28, 2004.

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

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

The proposed change eliminates the TS reporting requirements to provide a monthly operating report of shutdown experience and operating statistics if the equivalent data is submitted using an industry electronic database. It also eliminates the TS reporting requirement for an annual occupational radiation exposure report, which provides information beyond that specified in NRC regulations. The proposed change involves no changes to plant systems or accident analyses. As such, the change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accidents or transients. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3—The proposed change does not involve a significant reduction in a margin of safety?

This is an administrative change to reporting requirements of plant operating information and occupational radiation exposure data, and has no effect on plant equipment, operating practices or safety analyses assumptions. For these reasons, the proposed change does not involve a significant reduction in the margin of safety.

Based upon the reasoning presented above, the requested change does not involve significant hazards consideration.

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

NRC Section Chief: Mary Jane Ross-Lee, Acting.

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

Date of amendment request: October 5, 2004.

Description of amendment request: The proposed amendments would revise Technical Specifications (TSs) 3/4.3.1, ''Reactor Trip System Instrumentation,'' and 3/4.3.2, ''Engineered Safety Feature Actuation System Instrumentation,'' to modify steam generator (SG) level allowable value setpoints. The proposed changes address recent generic issues involving new SG level uncertainty considerations and margins associated with Westinghouse-designed SGs.

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

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

No. The SG water level-low-low setpoint and allowable value have been revised to address Westinghouse Nuclear Safety Advisory Letter NSAL–03–9 and other considerations on steam generator water level uncertainties. The revised setpoint and allowable value calculations continues to follow the setpoint methodology previously approved for BVPS Unit No. 1 and No. 2 while addressing newly identified level uncertainty considerations. The proposed changes to the SG water level-low-low Allowable Value for BVPS Unit No. 1 and No. 2 and to the SG water level-high-high Allowable Value for BVPS Unit No. 2 continue [to] maintain the validity of the safety analysis limits used in the safety analyses that credit the actuations based on SG water level.

The proposed changes do not alter the causes for any accident described in the Updated Final Safety Analysis Report (UFSAR) that credit the SG water level setpoint actuations. Therefore, they do not involve a significant increase in the probability of an accident previously evaluated.

The proposed changes do not alter the accident analyses that credit the SG water level-low-low setpoint actuation or the associated accident acceptance criteria. Therefore, they do not involve a significant increase in the consequences of an accident previously evaluated.

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

No. The SG water level-low-low setpoint and allowable value have been revised to address Westinghouse Nuclear Safety Advisory letter NSAL–03–9 and other considerations on steam generator water level uncertainties. Implementation of the proposed setpoint changes have no significant effect on either the configuration of the plant, or the manner in which the plant is operated. The proposed changes to the SG water level-low-low allowable value for BVPS Unit No. 1 and No. 2 and to the SG water level-high-high allowable value for BVPS Unit No. 2 continue to maintain the validity of the safety analysis limits used in the safety analyses that credit the actuations based on SG water level.

Therefore, since the plant configuration is not adversely changed and the proposed changes do not alter the accident analyses that credit actuation based on SG water level, the proposed change does not create the possibility of a new or different [kind of] accident from any accident previously evaluated.

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

No. The Reactor Trip System and Engineered Safety Feature Actuation System setpoint analysis methodology and acceptance criteria provide the margin of safety. The SG water level-low-low and SG water level-high-high actuation setpoint and allowable value have been calculated using the same methodology as previously approved for the BVPS Unit No. 1 and No. 2 while addressing newly identified considerations needed to protect the limits used in the safety analyses. The applicable safety analyses have been performed and show acceptable results. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mary O'Reilly, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: October 25, 2004.

Description of amendment request: The proposed amendment would revise Technical Specification 2.1.1.2 for the dual recirculation loop and single recirculation loop Safety Limit Minimum Critical Power Ratio (SLMCPR) values to reflect results of a cycle specific calculation.

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

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

Response: No.

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. Changing the SLMCPR does not increase the probability of an evaluated accident. The change does not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. Therefore, no individual precursors of an accident are affected.

The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits have been established, consistent with NRC approved methods, to ensure that fuel performance during normal, transient, and accident conditions is acceptable. The proposed change conservatively establishes the safety limit for the minimum critical power ratio (SLMCPR) for Cooper Nuclear Station Cycle 23 such that the fuel is protected during normal operation and during any plant transients or anticipated operational occurrences.

The proposed change revises the SLMCPR to protect the fuel during normal operation as well as during any transients or anticipated operational occurrences. Operational limits Minimum Critical Power Ratio (MCPR) are established based on the proposed SLMCPR to ensure that the SLMCPR is not violated during all modes of operation. This will ensure that the fuel design safety criteria (*i.e.*, that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and anticipated operational occurrences) is met. Since the operability of plant systems designed to mitigate any consequences of accidents has not changed, the consequences of an accident previously evaluated are not expected to increase.

Based on the above NPPD [Nebraska Public Power District] concludes that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

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. New accident precursors may be created by modifications of the plant configuration or changes in allowable modes of operation. The proposed change does not involve any modifications of the plant configuration or allowable modes of operation. The proposed change to the SLMCPR assures that safety criteria are maintained for Cycle 23.

Based on the above NPPD concludes that the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The value of the proposed SLMCPR provides a margin of safety by ensuring that no more than 0.1% of the rods are expected to be in boiling transition if the MCPR limit is not violated. The proposed change will ensure the appropriate level of fuel protection is maintained. Additionally, operational limits are established based on the proposed SLMCPR to ensure that the SLMCPR is not violated during all modes of operation. This will ensure that the fuel design safety criteria (*i.e.*, that at least 99.9% of the fuel rods do not experience transition boiling during normal operation as well as anticipated operational occurrences) are met.

Based on the above NPPD concludes that 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. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Acting Section Chief: Michael K. Webb.

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

Date of amendment request: April 15, 2004.

Description of amendment request: The proposed change would modify the Salem Updated Final Safety Analysis Report (UFSAR) with respect to fire protection requirements for the 4160 Volt Switchgear Rooms, 460 Volt Switchgear Rooms, and the Lower Electrical Penetration Area Rooms. Specifically, the amendment would reduce the UFSAR description of the Carbon Dioxide Tank volume from being able to provide two full discharges to an affected room to one full and one partial discharge to an affected room. Additionally, the assumed ability of the Carbon Dioxide system would be reduced from an ability to produce a CO_2 concentration of 50% for 30 minutes to an ability to produce a CO_2 concentration of 27.6% for a length of time sufficient to suppress a fire and allow the PSEG Nuclear Fire Department to respond.

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

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

Response: No.

The likelihood of a fire event is not increased since the proposed change does not alter the fire hazards contained in the plant. The ability to achieve and maintain safe shutdown in the event of a fire is not impacted by the reduction of CO_2 concentration, since the Fire Brigade will respond in ample time and extinguish a fire using alternate means. In addition, the proposed changes to the UFSAR would not change any response to a fire event. Also, the probability of occurrence or the consequences for an accident or malfunction of equipment is not increased by the proposed changes since the response to a fire event would not change and the fire brigade would continue to respond rapidly to any fires or fire alarms. Further, the proposed changes do not alter the way any structure, system, or component (SSC) functions, do not modify the manner in which the plant is operated, and do not significantly alter equipment out-of-service time. Changing the CO_2 concentration requirement in the 4160 Volt Switchgear Rooms, 460 Volt Switchgear Rooms and Lower Penetration Area Rooms at Salem Units 1 and 2 does not change the probability or consequences of any accident and dose consequences are unaffected. No changes to the design of structures, systems, or components (SSC) are made and there are no effects on accident mitigation.

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

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

Response: No.

The possibility of a new or different kind of accident from any accident or malfunction in the Salem Updated Final Safety Analysis Report (UFSAR) is not created. The design basis event applicable to the proposed change is a fire in the 4160 Volt Switchgear Rooms, 460 Volt Switchgear Rooms and Lower Penetration Area Rooms at Salem Units 1 and 2. Therefore a different accident is not created. In addition, the proposed changes cannot initiate an accident. Further, the proposed changes to the UFSAR do not change the design function or operation of any SSCs.

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

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

Response: No.

The reduction in CO_2 concentration provides ample response time for the onsite dedicated fire brigade to respond to a fire event and a 20% safety factor in CO_2 concentration remains. The proposed changes do not affect the ability to safely shutdown and maintain the shutdown conditions of either unit following a fire in the affected areas. The proposed changes do not rely on compensatory measures or actions deviating from the licensing or design basis. In addition, the proposed changes do not change the margin of safety since no SSCs are changed. The results of accident analysis remain unchanged by the proposed changes to the UFSAR.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of application request: September 17, 2004.

Description of amendment request: The amendment is to support the replacement of the steam generators (SGs) at Callaway during the refueling outage in the Fall of 2005. The amendment would (1) change the affected technical specifications (TSs) such as the reactor core safety limits (TS 2.1.1), reactor trip system (RTS) and engineered safety feature actuation system (ESFAS) instrumentation (TSs 3.3.1 and 3.3.2), reactor coolant system (RCS) limits (TS 3.4.1), RCS loops (TSs 3.4.5, 3.4.6, and 3.4.7), RCS operational leakage (TS 3.4.13), SG tube integrity (new TS 3.4.17), main steam safety valves (TS 3.7.1), SG surveillance program (TS 5.5.9), containment integrated leakage rate testing (ILRT) program (TS 5.5.16), and SG inspection report (TS 5.6.10); (2) revise the affected transient analyses such as excessive increase in secondary steam flow event, loss of normal feedwater event, transient mass and energy releases, radiological consequences of associated events, and containment pressure/temperature responses; and (3) revise nuclear steam and supply system (NSSS) design parameters and transients, and fatigue usage factors and stresses for the replacement SGs. The amendment involves the following areas of change to the license: nuclear steam supply system evaluations for the replacement steam generators, trip time delay (TTD) elimination for certain RTS and ESFAS functions, the SG surveillance program in Technical Specification Task Force (TSTF) No. 449 (TSTF–449), and the post-modification containment ILRT exception.

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 for the above areas of review, which is presented below (with the terms defined in the plant Technical Specifications capitalized):

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

Nuclear Steam Supply System Evaluations for Replacement Steam Generators

As discussed in the NSSS Licensing Report (Appendix A to this amendment application), all acceptance criteria continue to be met. All major NSSS components (*e.g.*, Reactor Vessel, Pressurizer, RCPs [(reactor coolant pumps)], Steam Generators, etc.) have been assessed with respect to bounding conditions expected for replacement steam generator (RSG) conditions. In all cases operation has been found to be acceptable. Major systems and subsystems (*e.g.*, safety injection, RHR [residual heat removal], etc.) have been reviewed and acceptable performance has been verified for their normal operation and, as applicable, for their safety-related functions. All reactor trip and ESFAS actuation setpoints have been assessed, and the proposed setpoint modifications will assure adequate protection is afforded for all design basis events.

The reactor core safety limits have been revised based on the RSG project parameters. All of the acceptance criteria for the accident analyses (*e.g.*, DNBR [departure from nucleate boiling ratio] limits, fuel centerline temperatures, etc.) continue to be met with the revised safety limit lines. Therefore, the revised core safety limit line changes are acceptable. The proposed changes to the reactor core safety limits will not initiate any accidents; therefore, they do not increase the probability of an accident previously evaluated in the FSAR [Callaway Final Safety Analysis Report]. The comprehensive analytical efforts performed to support the proposed RSG conditions include a reanalysis or evaluation of all accident analyses that are impacted by the revised reactor core safety limits.

The changes in various SG-related RTS and ESFAS Allowable Values have resulted from the analyses performed to support plant operation at the proposed RSG conditions. Setpoint uncertainty calculations confirm the acceptability of these revised Allowable Values. The affected RTS and ESFAS Allowable Values have been modified to reflect the results of updated setpoint calculations based on plant-specific uncertainties, calibration practices, calibration equipment, and installed hardware and procedures. The Allowable Values were calculated using the same Westinghouse setpoint methodology used for the current trip setpoints, but improved in a conservative fashion to include refinements that better reflect plant calibration practices and equipment performance. These refinements include the incorporation of a sensor reference accuracy term to address repeatability effects when performing a single pass calibration (*i.e.*, one up and one down pass at several points verifies linearity and hysteresis, but not repeatability). In addition, sensor and rack error terms for calibration accuracy and drift are grouped in the Channel Statistical Allowance equation with their dependent measurement and test equipment (M&TE) terms, then combined with the other independent error terms using the square root sum of the squares (SRSS) methodology. This improved setpoint methodology has been previously review[ed] and approved by the NRC. The proposed RTS and ESFAS Allowable Value changes will not initiate any accidents; therefore, they do not increase the probability of an accident previously evaluated in the FSAR. The comprehensive analytical effort performed to support the proposed RSG conditions included a reanalysis or evaluation of all accident analyses that are impacted by the revised RTS and ESFAS Allowable Values. All systems will function as designed.

The decrease in the Maximum Allowable Power for 3 OPERABLE MSSVs [main steam safety valves] per SG from $<$ 49% of Rated Thermal Power to $<$ 45% of Rated Thermal Power resulted from the analyses and evaluations performed to support plant operation at the proposed RSG conditions. The accident analysis acceptance criteria continue to be met with these changes. These proposed plant system changes do not increase the probability of an accident previously evaluated in the FSAR. The comprehensive analytical effort performed to support the proposed RSG conditions has included a review and evaluation of all components and systems (including interface systems and control systems) that could be affected by this change. All systems will function as designed. The change in the manner in which the Reactor Coolant Flow—Low Allowable Value is defined (while retaining the same numerical value), the change in the manner in which RCS average temperature is defined and the reduced upper limit for nominal T-avg [average temperature] at full power conditions in the Overtemperature ΔT [delta temperature] and Overpower ΔT setpoint equations, and the changes to the pressurizer pressure and RCS average temperature limits in the DNB LCO [departure from nucleate boiling limiting condition for operation] [TS] 3.4.1 have also been evaluated. None of these proposed changes will initiate any accidents; therefore, the probability of an accident has not been increased.

The potential dose consequences have been analyzed with respect to the above changes collectively. The dose increases are less than minimal (*i.e.*, <10% of the margin between the regulatory limits and the currently reported doses). The applicable dose acceptance criteria continue to be met.

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

Trip Time Delay Elimination

This design change will eliminate only the Trip Time Delay portion of the SG Water Level Low-Low trip functions and return that portion of the design to condition that existed prior to Callaway Amendment 43 dated April 14, 1989. The coincidence logic in the Solid State Protection System will be unaffected. In all other regards, the design of the RTS and ESFAS instrumentation will be unaffected. These protection systems will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to this amendment request are maintained.

The probability and consequences of accidents previously evaluated in the FSAR are not adversely affected because the removal of the trip time delay circuitry assures a faster response by the affected trip functions, consistent with the safety analysis acceptance criteria and the original plant licensing basis.

The proposed change will not affect the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance.

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

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

TSTF–449 Generic Licensing Change Package

This proposed change requires a Steam Generator Program that includes performance criteria that will provide reasonable assurance that the steam generator (SG) tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown, and all anticipated transients included in the design specification). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational LEAKAGE.

A steam generator tube rupture (SGTR) event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. In the analysis cases for the SGTR event at Callaway Plant, a primary to secondary LEAKAGE rate of 1 gallon per minute (gpm) to the unaffected SGs is assumed, in excess of the RCS Operational LEAKAGE rate limit in TS 3.4.13, and the LEAKAGE rate associated with a double-ended rupture of a single tube in the ruptured SG is also assumed. For other design basis accidents such as main steam line break (MSLB), rod ejection, and reactor coolant pump locked rotor, the SG tubes are assumed to retain their structural integrity (*i.e.*, they are assumed not to rupture). These additional analyses for Callaway Plant assume, as an initial condition, that primary to secondary LEAKAGE for all SGs is 1 gpm. The accident induced leakage criterion introduced by the proposed change to TS 5.5.9 accounts for tubes that may leak during design basis accidents. The accident induced leakage criterion limits this leakage to no more than the 1 gpm value assumed in the accident analyses.

The SG performance criteria added to TS 5.5.9 identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the Steam Generator Program required by the proposed change to TS 5.5.9. The program, defined by NEI [Nuclear Energy Institute] 97–06, Steam Generator Program Guidelines, includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring.

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT I–131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in TS 3.4.13 for RCS Operational leakage and in TS 3.4.16 for DOSE EQUIVALENT I–131 in the primary coolant to ensure the plant is operated within its analyzed condition. The radiological consequence analyses at Callaway Plant assume that the primary to secondary LEAKAGE rate is 1 gpm (more conservative than the limit in TS 3.4.13), and that the reactor coolant activity levels of DOSE EQUIVALENT I–131 are at the TS 3.4.16 limits.

The proposed TSTF–449 changes reflect the design of the replacement SGs, but do not affect their method of operation or primary or secondary coolant chemistry controls. The proposed changes update the TS and enhance the requirements for SG inspections. The proposed changes do not adversely impact the conclusions of any previously evaluated design basis accident and are an improvement over the existing TS.

Therefore, this proposed change to implement TSTF–449 does not affect the consequences of a SGTR accident and the probability of such an accident is reduced. In addition, this proposed change does not affect the consequences of an MSLB, rod ejection, reactor coolant pump locked rotor, or any other accident event involving the potential release of radioactive fluids from the secondary side of Callaway Plant. [Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.]

Post-Modification ILRT Exception

This proposed change would provide Callaway Plant with an exception from performing a post-modification containment integrated leak rate test following the replacement of the steam generators during Refuel [Outage] 14.

Integrated leak rate tests are performed to assure the leak-tightness of the primary containment boundary system, and as such they are not accident initiators. Therefore, not performing an integrated leak rate test will not affect the probability of an accident previously evaluated. The intent of post-modification integrated leak rate testing requirements is to assure the leak-tight integrity of the area affected by the modification. For the Callaway Plant steam generator replacement modification, this intent will be satisfied by performing the American Society of Mechanical Engineers code required inspections and tests. Since the leak-tightness integrity of the primary containment boundary affected by the steam generator replacement will be assured, there is no change in the containment boundary's ability to confine radioactive materials during an accident. Therefore, adding a Technical Specification exception from the steam generator replacement post-modification integrated leak rate testing requirements does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Nuclear Steam Supply System Evaluations for Replacement Steam Generators

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

This amendment does not alter the safe performance of the plant protection systems to trip the reactor when necessary or actuate ESF [engineered safety feature] systems.

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

Trip Time Delay Elimination

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

This amendment does not alter the safe performance of the plant protection systems to trip the reactor when necessary or actuate ESF systems.

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

TSTF–449 Generic Licensing Change Package

The proposed performance based requirements are an improvement over the requirements imposed by the existing TS.

Implementation of the proposed Steam Generator Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The result of the implementation of the Steam Generator Program will be an enhancement of SG tube performance. Primary to secondary LEAKAGE that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

This proposed change does not impact the method of SG operation or primary or secondary coolant chemistry controls. In addition, this proposed change does not impact any other plant system or component. The change enhances SG inspection requirements.

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

Post-Modification ILRT Exception

The proposed change would provide Callaway Plant with an exception from performing a post-modification containment integrated leak rate test following the replacement of the steam generators during Refuel 14. Providing an exception from performing a test does not involve a physical change to the plant nor does it change the operation of the plant. Thus it cannot introduce a new failure mode. Therefore adding a Technical Specification requirement that provides an exception from the steam generator replacement post-modification integrated leak rate testing requirement does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Nuclear Steam Supply System Evaluations for Replacement Steam Generators

The analyses and evaluations supporting the proposed RSG conditions reflect the reactor core safety limits. All acceptance criteria continue to be met.

The analyses supporting the proposed RSG conditions reflect the proposed RTS and ESFAS Allowable Values. Setpoint calculations demonstrate that margin exists between these Allowable Values and the corresponding safety analysis limits used in the RSG analyses. The calculations are based on plant instrumentation and calibration/functional test methods and include allowances for the RSG conditions. All analyses and evaluations supporting the proposed RSG core safety limits, decrease in maximum allowable power level for 3 operable MSSVs per SG, the change in the manner in which the Reactor Coolant Flow—Low Allowable Value is defined (while retaining the same numerical value), the change in the manner in which RCS average temperature is defined and the reduced upper limit for nominal T-avg at full power conditions in the Overtemperature ΔT and Overpower ΔT setpoint equations, and the changes to the pressurizer pressure and RCS average temperature limits in the DNB LCO [TS] 3.4.1 are acceptable. All acceptance criteria continue to be met. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Trip Time Delay Elimination

This proposed change does not eliminate any RTS or ESFAS surveillances or alter the frequency of those surveillances as required by the TS. The SG Water Level Low—Low safety analysis limit of 0% span assumed in the analyses supporting the approval of the TTD design in Callaway Amendment 43 dated April 14, 1989 is also used in the RSG analyses discussed above. None of the acceptance criteria for any accident analysis is changed for TTD elimination.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. The radiological dose consequence acceptance criteria will continue to be met.

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

TSTF–449 Generic Licensing Change Package

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a SG is maintained by ensuring the integrity of its tubes.

Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. This proposed change to implement TSTF–449 does not, of itself, affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the Steam Generator Program to manage SG tube inspection, assessment, repair (only under NRC-approved methods, none of which currently apply to the RSGs), and plugging. The requirements established by the Steam Generator Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the existing TS.

For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by this proposed change.

Post-Modification ILRT Exception

The proposed change would provide Callaway Plant with an exception from performing a post-modification containment integrated leak rate test following the replacement of the steam generators during Refuel 14. The intent of post-modification integrated leak rate testing requirements is to assure the leak-tight integrity of the area affected by the modification. This intent will be satisfied by performing American Society of Mechanical Engineers code required inspections and tests. The acceptance criterion for American Society of Mechanical Engineers code system pressure testing for the base metal and welds is no leakage. In addition, the test pressure for the system pressure test will be several times that required during an integrated leak rate test. Since the leak-tight integrity of the primary containment boundary affected by the steam generator replacement will be assured, there is no change in the primary containment boundary's ability to confine radioactive materials during an accident. Therefore, adding a Technical Specification requirement that provides an exception from the steam generator replacement post-modification integrated leak rate testing requirements does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: September 15, 2004.

Description of amendment request: The proposed changes will change the Administrative Controls Section of the Technical Specifications (TS) in order to incorporate title changes, change the location where the plant-specific titles and TS titles are correlated, and relocate the unit staff requirements to the Quality Assurance Program. These proposed changes will support the implementation of proposed Virginia Electric and Power Company Topical Report DOM–QA–1, ''Nuclear Facility Quality Assurance Program Description,'' currently under U.S. Nuclear Regulatory Commission (NRC) staff review.

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

1. Operation of North Anna Units 1 and 2 in accordance with the proposed license amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change is administrative in nature and does not affect plant systems, structures or components (SSCs) or plant operation during normal or accident conditions. The proposed change only affects the designated titles of personnel, the location of the TS title and plant-specific title correlation, and the location of the unit staff qualification requirements. Therefore, this change has no bearing on the probability of an accident. Management organizational structure and safety and operational reviews have not changed and there is no change in the method of plant operation, operation review, or system design review. As such, this change does not alter the conclusions of the existing safety analyses and therefore does not alter the consequences of an accident previously evaluated.

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

The proposed administrative change continues to ensure that adequate management oversight exists at the plant in accordance with the existing Technical

Specifications. The proposed change only affects the designated titles of personnel, the location of the TS title and plant-specific title correlation, and the location of the unit staff qualification requirements. This change does not impact plant SSCs or plant operation. Management organizational structure and safety and operational reviews have not changed and there is no change in the method of plant operation, operation review, or system design review. There are no new or different accident scenarios, accident initiators, nor failure mechanisms that will be introduced due to this change. Therefore, the proposed change does not create the possibility of an accident of a different type than evaluated previously.

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

The proposed change only affects the designated titles of personnel, the location of the TS title and plant-specific title correlation, and the location of the unit staff qualification requirements. This change does not impact plant design, plant operation or any safety margin. Therefore, the proposed change does not significantly reduce a margin of safety.

This evaluation concludes that the proposed amendments to the North Anna Units 1 and 2 Technical Specifications do not involve a significant increase in the probability or consequences of a previously evaluated accident, do not create the possibility of a new or different kind of accident and do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Ms. Lillian M. Cuoco, Esq., Senior Counsel, Dominion Resources Services, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385.

NRC Section Chief: Mary Jane Ross-Lee (Acting).

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

Date of amendment request: October 7, 2004.

Description of amendment request: The proposed change would revise Technical Specification (TS) 5.3, ''Unit Staff Qualifications,'' to reinstate the qualification requirements for the shift manager and control room supervisor positions that were inadvertently eliminated through Amendment No. 150. Also, TS 5.3 would be revised to reference this amendment application for the use of the National Academy for Nuclear Training guideline, ACAD 00–003, Revision 1, ''Guidelines for Initial Training and Qualification of Licensed Operators.'' Various other TSs would be revised to make corrections that were identified by the NRC staff in its letter dated January 28, 2004, and additional reviews performed by the licensee.

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

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

Unit Staff Qualifications

The proposed change is an administrative change to reinstate the qualification requirements for specific control room positions that were inadvertently eliminated through the issuance of Amendment No. 150 and utilize Revision 1 to ACAD 00–003, ''Guidelines for Initial Training and Qualification of Licensed Operators.'' The proposed change does not directly impact accidents previously evaluated. WCNOC's [Wolf Creek Nuclear Operating Corporation's] licensed operator training program is accredited by the National Academy for Nuclear Training and is based on a systems approach to training consistent with the requirements of 10 CFR 55. Although licensed operator qualifications and training may have an indirect impact on accidents previously evaluated, the NRC considered this impact during the rulemaking process, and by promulgation of the revised 10 CFR 55 rule, concluded that this impact remains acceptable as long as the licensed operator training program is certified to be accredited and is based on a systems approach to training.

Corrections

The proposed change involves corrections to the Technical Specifications that are either associated with the issuance of the Improved Technical Specifications (Amendment No. 123) or subsequent amendments. The changes are considered administrative changes and do not modify, add, delete, or relocate any technical requirements of the Technical Specifications. As such, administrative changes do not effect initiators of analyzed events or assumed mitigation of accident or transient events.

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

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

Unit Staff Qualifications

The proposed change is an administrative change to reinstate the current requirements of specific control room positions and allow the use of Revision 1 of ACAD 00–003 for initial training and qualification of licensed operators. WCNOC's licensed operator training program is accredited by the National Academy for Nuclear Training and is based on a systems approach to training consistent with the requirements of 10 CFR 55. Although licensed operator qualifications and training may have an indirect impact on accidents previously evaluated, the NRC considered this impact during the rulemaking process, and by promulgation of the revised 10 CFR 55 rule, concluded that this impact remains acceptable as long as the licensed operator training program is certified to be accredited and is based on a systems approach to training.

Corrections

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

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

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

Unit Staff Qualifications

The proposed change is an administrative change to reinstate the current requirements of specific control room positions and allow the use of Revision 1 of ACAD 00–003 for initial training and qualification of licensed operators. As noted previously, WCNOC's licensed operator training program is accredited and is based on a systems approach to training consistent with the requirements of 10 CFR 55. Licensed operator qualifications and training can have an indirect impact on the margin of safety. However, the NRC considered this impact during the rulemaking process, and by promulgation of the revised 10 CFR 55 rule, determined that this impact remains acceptable when licensees maintain a licensed operator training program that is accredited and based on a systems approach to training.

Corrections

The proposed change will not reduce a margin of safety because they have no effect on any safety analysis assumptions. The change is administrative in nature.

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

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

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

NRC Section Chief: Robert Gramm.

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.

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

Date of amendment request: October 22, 2004.

Description of amendment request: The proposed amendment would revise the allowed outage times of Technical Specification 3.3.3.6, ''Accident Monitoring Instrumentation,'' to be consistent with the completion times in the related specification in NUREG–1431, Revision 3, ''Standard Technical Specifications Westinghouse Plants.''

*Date of publication of individual notice in **Federal Register:*** November 2, 2004 (69 FR 63560).

Expiration date of individual notice: December 2, 2004 (public comments) and January 3, 2005 (hearing requests).

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by email to *pdr@nrc.gov.* (**Note:** Public access to ADAMS has been temporarily suspended so that security reviews of publicly available documents may be performed and potentially sensitive information removed. Please check the NRC Web site for updates on the resumption of ADAMS access.)

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

Date of application for amendment: December 23, 2003, as supplemented by letter dated June 16, 2004.

Brief description of amendment: The amendment revised Section 4.5.D of the Technical Specifications to specify testing the main steam isolation valves at a pressure lower than Pa, the calculated peak containment internal pressure related to the design-basis loss-of-coolant accident.

Date of Issuance: November 2, 2004.

Effective date: November 2, 2004 and shall be implemented within 30 days of issuance

Amendment No.: 250.

Facility Operating License No. DPR–16: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 17, 2004 (69 FR 7518).

The June 16, 2004, letter provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated November 2, 2004.

No significant hazards consideration comments received: No.

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

Date of application of amendments: January 15, 2004, as supplemented by letter dated March 15, 2004.

Brief description of amendments: The amendments revise the Technical Specifications associated with the control rod drive trip devices. The amendments are needed to support implementation of the reactor trip breaker replacement.

Date of Issuance: November 2, 2004.

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

Amendment Nos.: 341, 343, 342.

Renewed Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** April 13, 2004 (69 FR 19566). The supplement dated March 15, 2004, provided clarifying information that did not change the scope of the January 15, 2004, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 2, 2004.

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 (BVPS–1 and 2), Beaver County, Pennsylvania

Date of application for amendments: March 22, 2004, as supplemented July 23 and October 11, 2004.

Brief description of amendments: The amendments modified Technical Specification (TS) requirements to adopt the provisions of Industry/TS Task

Force (TSTF) change TSTF–359, ''Increased Flexibility in Mode Restraints.''

Date of issuance: November 4, 2004.

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

Amendment Nos.: 263 and 144.

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

*Date of initial notice in **Federal Register:*** August 31, 2004 (69 FR 53108). The supplemental letters dated July 23 and October 11, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 4, 2004.

No significant hazards consideration comments received: No.

Nine Mile Point Nuclear Station, LLC, Docket Nos. 50–220, and 50–410, Nine Mile Point Nuclear Station, Unit Nos. 1 and 2, Oswego County, New York

Date of application for amendments: January 8, 2004 (2 letters), as supplemented by letter dated June 17, 2004.

Brief description of amendments: The amendments approve implementation of the Boiling Water Reactor Vessel and Internals Project Reactor Pressure Vessel Integrated Surveillance Program as the basis for demonstrating the units' compliance with the requirements of appendix H to Title 10 of the Code of Federal Regulations. Specifically, the amendments approved the wording proposed by the licensee to update the units' Updated Safety Analysis Reports. In addition, the Unit 1 amendment also revised the Technical Specifications to delete any reference to plant-specific surveillance requirements.

Date of issuance: November 8, 2004.

Effective date: As of the date of issuance. Integrated Surveillance Program shall be implemented within 90 days of issuance. The units' Final Safety Analysis Report (Updated) shall be updated in accordance with 10 CFR 50.71(e).

Amendment Nos.: 184 and 114.

Facility Operating License Nos. DPR–63 and NPF–69: Amendments revise the Technical Specifications (for Unit 1), the operating license (for Unit 2), and approve revision of licensing basis for both units.

*Date of initial notice in **Federal Register:*** February 17, 2004 (69 FR 7524). The June 17, 2004, letter provided clarifying information within the scope of the original application 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 two Safety Evaluations, both dated November 8, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 23, 2003, as supplemented June 21, 2004.

Brief description of amendment: The amendment changes Technical Specification (TS) Limiting Condition for Operation (LCO) Tables 3.2.1 and 3.2.4 to (1) eliminate the reactor head cooling containment isolation function from the TSs, (2) correct and clarify the description of the number of instrument channels per trip system as defined in the TSs, and (3) revise an existing LCO for radiation monitors used to isolate reactor building ventilation and initiate the standby gas treatment system.

Date of issuance: November 2, 2004.

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

Amendment No.: 140.

Facility Operating License No. DPR–22. Amendment revised the TSs.

*Date of initial notice in **Federal Register:*** March 30, 2004 (69 FR 16621).

The supplemental letter 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 November 2, 2004.

No significant hazards consideration comments received: No.

Pacific Gas and Electric Company, Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of application for amendments: March 18, 2004, and its supplements dated August 18 and 20, and September 17, 2004.

Brief description of amendments: The amendments authorize revisions to the Final Safety Analysis Report (FSAR) Update to incorporate the NRC approval of a permanently revised steam generator voltage-based repair criteria probability of detection (POD) method. The revised POD method is referred to as the probability of prior cycle detection method. In addition, a reporting requirement is added to the DCPP Technical Specifications as TS 5.6.10.i.

Date of issuance: October 28, 2004.

Effective date: October 28, 2004, and shall be implemented within 30 days of the date of issuance. The implementation of the amendment includes the incorporation into the FSAR Update the changes discussed above, as described in the licensee's application dated March 18, 2004, and its supplements dated August 18 and 20, and September 17, 2004, and evaluated in the staff's Safety Evaluation attached to the amendments.

Amendment Nos.: Unit 1–177; Unit 2–179.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the FSAR Update and the Technical Specifications.

*Date of initial notice in **Federal Register:*** June 22, 2004 (69 FR 34704).

The August 18 and 20, and September 17, 2004, supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 28, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 22, 2003, as supplemented by letters dated June 18, July 15, and September 8, 2004.

Brief description of amendments: The amendment added TS 3.3.1.3, ''Oscillation Power Range Monitor (OPRM) Instrumentation,'' and changed TS 3.4.1, ''Recirculation Loops Operating,'' and TS 5.6.5, ''Core Operating Limits Report,'' to remove specifications and information related to current stability specifications which will no longer be needed with the operation of the OPRM system.

Date of issuance: November 9, 2004.

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

Amendment Nos.: 217 and 192.

Facility Operating License Nos. NPF–14 and NPF–22: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 20, 2004 (69 FR 2745). The supplements dated June 18, July 15, and September 8, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 9, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 31, 2003, as supplemented by letter dated July 30, 2004.

Brief description of amendment: The amendment revised the reactor pressure vessel pressure-temperature limits and extends their validity to 32 effective full power years.

Date of issuance: November 1, 2004.

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

Amendment No.: 157.

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

Date of initial notice in ***Federal Register:*** June 8, 2004 (69 FR 32076). The July 30, 2004 letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 1, 2004.

No significant hazards consideration comments received: No.

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

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

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

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

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

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

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

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. Within 60 days after the date of publication of this notice, 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 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, and electronically on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If there are problems in accessing the document, contact the PDR Reference staff at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov*. (**Note:** Public access to ADAMS has been temporarily suspended so that security reviews of publicly available documents may be performed and potentially sensitive information removed. Please check the NRC Web site for updates on the resumption of ADAMS access.) If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.309, 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 general requirements: (1) The name, address and telephone number of the requestor or petitioner; (2) the nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; (3) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and (4) the possible effect of any decision or order which may be entered in the proceeding on the requestor's/petitioner's interest. The petition must also identify the specific contentions which the petitioner/requestor seeks to have litigated at the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner/requestor shall provide a brief explanation of the bases for 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. The petition must include sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact.[1] 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/requestor who fails to satisfy these requirements with respect to at least one contention will not be permitted to participate as a party.

Each contention shall be given a separate numeric or alpha designation within one of the following groups:

1. Technical—primarily concerns/issues relating to technical and/or health and safety matters discussed or referenced in the applications.
2. Environmental—primarily concerns/issues relating to matters discussed or referenced in the environmental analysis for the applications.
3. Miscellaneous—does not fall into one of the categories outlined above.

As specified in 10 CFR 2.309, if two or more petitioners/requestors seek to co-sponsor a contention, the petitioners/requestors shall jointly designate a representative who shall have the authority to act for the petitioners/requestors with respect to that contention. If a petitioner/requestor seeks to adopt the contention of another sponsoring petitioner/requestor, the petitioner/requestor who seeks to adopt the contention must either agree that the sponsoring petitioner/requestor shall act as the representative with respect to that contention, or jointly designate with the sponsoring petitioner/requestor a representative who shall have the authority to act for the petitioners/requestors with respect to that contention.

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. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed by: (1) First class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff; (2) courier, express mail, and expedited delivery services: Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852, Attention: Rulemaking and Adjudications Staff; (3) E-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, *hearingdocket@nrc.gov;* or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC, Attention: Rulemakings and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

Nontimely requests and/or petitions and contentions will not be entertained absent a determination by the Commission or the presiding officer or the Atomic Safety and Licensing Board that the petition, request and/or the contentions should be granted based on a balancing of the factors specified in 10 CFR 2.309(a)(1)(i)–(viii).

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

Date of amendment request: September 30, 2004.

Description of amendment request: The amendment changes Technical Specification 4.4.4.2 to expand the range of conditions under which quarterly testing of block valves for the pressurizer power operated relief valves would be unnecessary.

Date of issuance: October 21, 2004.

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

Amendment No.: 153.

[1] To the extent that the applications contain attachments and supporting documents that are not publicly available because they are asserted to contain safeguards or proprietary information, petitioners desiring access to this information should contact the applicant or applicant's counsel and discuss the need for a protective order.

Facility Operating License No. NPF–80: Amendment revises the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. October 6, 2004 (69 FR 59969). The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided an opportunity to request a hearing by December 6, 2004, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated October 21, 2004.

Attorney for licensee: Mr. John E. Matthews, Morgan, Lewis & Bokius, LLP, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Section Chief: Michael K. Webb, Acting.

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

Date of amendment request: November 2, 2004.

Description of amendment request: The amendment revises Technical Specification Limiting Condition for Operation 3.4.3, ''Primary Coolant System (PCS) Pressure and Temperature (P/T) Limits'' to add restrictions to the cooldown rate limits. This amendment supports plant restart following repairs of two reactor vessel closure head control rod drive nozzle penetrations at the Palisades Nuclear Power Plant.

Date of issuance: November 8, 2004.

Effective date: As of the date of issuance and shall be implemented immediately.

Amendment No.: 218.

Facility Operating License No. DPR–20: Amendment revises the Technical Specification.

Public comments requested as to proposed no significant hazards consideration (NSHC):

The Commission's related evaluation of the amendment, finding of emergency circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated November 8, 2004.

Attorney for licensee: Jonathan Rogoff, Esquire, Vice President, Counsel & Secretary, Nuclear Management Company, LLC, 700 First Street, Hudson, WI 54016.

NRC Section Chief: L. Raghavan.

Tennessee Valley Authority, Docket No. 50–327, Sequoyah Nuclear Plant, Unit 1, Hamilton County, Tennessee

Date of amendment request: November 4, 2004.

Description of amendment request: The proposed amendment extended the implementation period for License Amendment 294 to May 15, 2005.

Date of issuance: November 9, 2004.

Effective date: As of date of issuance, to be implemented by May 15, 2005.

Amendment No.: 297.

Facility Operating License No. DPR–77: Amendment revises the implementation date for License Amendment No. 294.

Public comments requested as to proposed no significant hazards consideration (NSHC):

The Commission's related evaluation of the amendment, finding of emergency circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated November 9, 2004.

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

NRC Section Chief: Michael L. Marshall, Jr.

Dated at Rockville, Maryland, this 15th day of November, 2004.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. 04–25664 Filed 11–22–04; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

Excepted Service

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: This gives notice of OPM decisions granting authority to make appointments under Schedules A, B and C in the excepted service as required by 5 CFR 6.6 and 213.103.

FOR FURTHER INFORMATION CONTACT: Hughes Turner, Deputy Associate Director, Center for Leadership and Executive Resources Policy, Division for Strategic Human Resources Policy, 202–606–1811.

SUPPLEMENTARY INFORMATION: Appearing in the listing below is one Schedule A appointment, no Schedule B appointments, and Schedule C appointments established between October 1, 2004 and October 31, 2004. Future notices will be published on the fourth Tuesday of each month, or as soon as possible thereafter. A consolidated listing of all authorities as of June 30 is published each year.

Schedule A

Department of Homeland Security 213.3111

Up to 15 Senior Level and General Schedule (or equivalent) positions within the Homeland Security Labor Relations Board and the Homeland Security Mandatory Removal Panel. Effective October 15, 2004.

Schedule B

No Schedule B appointments for October 2004.

Schedule C

The following Schedule C appointments were approved for October 2004:

Section 213.3303 Executive Office of the President, Office of National Drug Control Policy

QQGS00083 Intergovernmental Affairs Liaison to the Chief of Staff. Effective October 19, 2004.

QQGS00086 Legislative Assistant to the Associate Director, Legislative Affairs. Effective October 19, 2004.

Section 213.3304 Department of State

DSGS60797 Legislative Management Officer to the Assistant Secretary for Legislative and Intergovernmental Affairs. Effective October 14, 2004.

DSGS60798 Legislative Management Officer to the Assistant Secretary for Legislative and Intergovernmental Affairs. Effective October 14, 2004.

DSGS60799 Foreign Affairs Officer to the Under Secretary for Global Affairs. Effective October 14, 2004.

DSGS60800 Staff Assistant to the Senior Advisor to the Secretary and White House Liaison. Effective October 28, 2004.

Section 213.3304 Department of Treasury

DYGS00434 Special Assistant to the Deputy Chief of Staff. Effective October 25, 2004.

Section 213.3306 Department of Defense

DDGS16831 Research Assistant to the Deputy Assistant Secretary of Defense (Strategic Communications Planning). Effective October 1, 2004.

DDGS16842 Staff Assistant to the Deputy Assistant Secretary of Defense (Special Operations and Combating Terrorism). Effective October 20, 2004.