NW., Washington, DC, and accessible electronically through the ADAMS Public Electronic Reading Room link at the NRC Web site (http://www.nrc.gov).

Dated at Rockville, Maryland this 23rd day of June 2000.

For the Nuclear Regulatory Commission.

George F. Wunder,

Project Manager, Section 1, Project Directorate 1, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

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NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from June 3, 2000, through June 16, 2000. The last biweekly notice was published on June 14, 2000 (65 FR 37420).

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

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

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

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

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

By July 28, 2000, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

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: June 14, 2000.

Description of amendment request: The requested amendment proposes to revise Technical Specification (TS) 5.6.5 to incorporate analytical methodologies that are used for core operating limits that have been accepted by NRC for referencing in licensing applications.

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:

Carolina Power & Light (CP&L) Company has evaluated the proposed TS change and has concluded that it does not involve a significant hazards consideration. The conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

The proposed changes in a methodology have been previously generically reviewed and approved for use by the NRC for determining core operating limits. Analyzed events are assumed to be initiated by the failure of plant structures, systems, or components. The core operating limits developed in accordance with the new methodologies are bounded by the limitations in the NRC acceptance in its safety evaluations of the new methodologies. The topical reports associated with the new methodologies demonstrate that the integrity of the fuel will be maintained during normal operations and that design requirements will continue to be met. The proposed change does not have a detrimental impact on the integrity of any plant structure, system, or component. The proposed change will not alter the operation of any plant equipment, or otherwise increase its failure probability. Therefore, the probability of occurrence for a previously analyzed accident is not significantly increased.

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

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

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

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

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

The margin of safety is established through the design of the plant structures, systems, and components, through the parameters within which the plant is operated, through the establishment of the setpoints for the actuation of equipment relied upon to respond to an event, and through margins contained within the safety analyses. The proposed change in the methodologies used for MSLB and LBLOCA analyses and the use of the generic design criteria for PWR [pressurized-water reactor] fuel designs does not impact the condition or performance of structures, systems, setpoints, and components relied upon for accident mitigation. The proposed change does not significantly, impact any safety analysis assumptions or results. Therefore, the proposed change does not result in a significant reduction in the margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: April 26, 2000.

Description of amendment request: The proposed amendments would revise Technical Specification Sections 3/4.3.7.1, ''Radiation Monitoring Instrumentation,'' 3/4.7.2, ''Control Room and Auxiliary Electric Equipment Room Emergency Filtration System,'' and 6.2.F.8, ''Ventilation Filter Testing Program,'' to eliminate habitability system requirements associated with the Auxiliary Electric Equipment Room habitability systems.

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:

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

The elimination of Auxiliary Electric Equipment Room (AEER) habitability system requirements does not affect the precursors or initiators of any accidents previously evaluated.

The current analysis assumes an operator will maintain continuous occupancy of the AEER for 30 days following a design basis loss-of-coolant-accident (LOCA). This analysis credits operation of the AEER habitability system. The resultant dose to the operator is within the limits of 10 CFR 50, Appendix A, ''General Design Criteria for Nuclear Power Plants,'' General Design Criterion (GDC) 19, ''Control Room.'' We have performed an evaluation that determined an operator has more than sufficient time to perform all required actions in the AEER following a design basis LOCA, when directed by the station's emergency operating procedures (EOPs), without taking credit for the AEER habitability system and still maintain the resultant dose within the limits of GDC 19.

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

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

The proposed changes do not effect the operation or configuration of plant systems, structures, or components. These proposed changes do not affect currently analyzed failure modes and do not introduce new failure modes.

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

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

The proposed changes will require an operator to be present in the AEER in a post-LOCA environment only when necessary to perform required actions as directed by the station's EOPs. A time/motion study of required AEER actions has determined that the maximum cumulative time spent in the AEER is approximately 300 minutes. The dose to operators performing the required AEER actions, without credit for the AEER filtration system, will continue to be within the limits of GDC 19, during and following all design basis accidents.

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

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

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

NRC Section Chief: Anthony J. Mendiola.

GPU Nuclear, Inc. and Saxton Nuclear Experimental Corporation, Docket No. 50–146, Saxton Nuclear Experimental Facility (SNEF), Bedford County, Pennsylvania

Date of amendment request: April 10, 2000.

Description of amendment request: The proposed amendment would make changes to the organizational and administrative controls for the SNEF to reflect changes in GPU Nuclear, Inc. following the sale of the Oyster Creek Nuclear Generating Station. The proposed changes to the technical specifications (TSs) would (1) replace reference to the President of GPU Nuclear and division Vice Presidents with a GPU Nuclear Cognizant Officer, (2) replace reference to ''other GPU Nuclear personnel'' with ''other GPU Inc, personnel,'' (3) replace reference to the ''Radiation Safety Committee'' with the ''TMI2/SNEC Oversight Committee,'' (4) replace ''GPU Nuclear audit program procedures'' with ''approved Quality Assurance Plan procedures,'' and (5) make changes to the TSs to reflect changes to NRC organization.

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

GPUN has determined that Technical Specifications Change request No. 60 involves no significant hazards consideration as defined in 10 CFR 50.92.

1. The proposed changes to the SNEC Technical Specifications do not involve a significant increase in the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously analyzed in the safety analysis report. The changes have no impact on plant operations or the release of radioactive materials.

2. The proposed changes to the SNEC Technical Specifications will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the safety analysis report because no plant configuration or operational changes are involved.

3. The changes will not involve a significant reduction in the margin of safety as defined in the basis for any technical specification for SNEC because no change to operational limits will be made.

The NRC staff has reviewed the analysis of the licensees 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 the Licensee: Ernest L. Blake, Jr., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Branch Director: Ledyard B. Marsh.

Northeast Nuclear Energy Company (NNECO), et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: February 3, 2000.

Description of amendment request: NNECO's proposed license amendment request of February 3, 2000, would add a note to the Millstone 3 Final Safety Analysis Report (FSAR) to indicate that the configuration of relief valve 3CHS*V62 and isolation valve 3CHS*V61 takes exception to American Society of Mechanical Engineers (ASME) Section III code requirements for class 2 components. The change does not affect existing plant design but rather changes licensing basis information in the FSAR to accurately reflect plant configuration.

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. The NRC staff reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below:

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

The revision to the Final Safety Analysis Report (FSAR) to correctly reflect the current valve configuration to the Chemical Volume and Control System (CVCS) will not affect the ability of the CVCS to perform its intended safety function. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Since there are no changes in components, component operation, or system operation, this change does not create the possibility of an accident of a different type.

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

Since the FSAR revision does not have anything to do with affecting the ability of the CVCS to perform its intended safety function, it will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

Northern States Power Company, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of amendment request: May 12, 2000.

Description of amendment request: The proposed amendment would revise the Technical Specification Section 4.6.E.1.d safety/relief valve (SRV) bellows monitoring system test frequency from quarterly to once per operating cycle.

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

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

The proposed amendment will have no impact on the probability or consequences of an accident. The BLDS [bellows leak detection system] performs a monitoring function only and is not part of the reactor pressure boundary.

The reduced testing frequency for the leak detection monitoring function will have no impact on the ability of the pressure switch to detect a bellows failure or on the likelihood of bellows failure. Experience has shown the pressure switch to be reliable and capable of performing its function.

Reduction in test frequency to once per cycle will still provide periodic verification of pressure switch capability. Reduction in test frequency to once per cycle will reduce the number of times per cycle that SRV operability is impacted by the testing process. This will increase the probability that SRV's [sic] would be available to mitigate consequences of an accident.

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

The proposed amendment has the potential to improve reliability of the BLDS by removing a requirement which will allow removal of a failure path. A reduction in BLDS surveillance test frequency will not result in creation of a new or different kind of accident. The BLDS performs a monitoring function only. It cannot cause an accident as it is not part of the reactor pressure boundary.

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

Revising the requirement to test this system from quarterly to once per cycle will not reduce the margin of safety. The pressure switch and pressure boundary components of the BLDS are reliable and stable. Therefore, the proposed Technical Specification change does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Claudia M. Craig.

Public Service Electric & Gas Company, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Units Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: March 2, 2000.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3/4.6.3, ''Containment Isolation Valves.'' The proposed change deletes the asterisk (*) modifying the word OPERABLE in the Limiting Condition for Operation and relocates its associated footnote at the bottom of the page to immediately following the Action Statement. The new note would be reworded to be consistent with the wording of NUREG–1431, ''Standard Technical Specifications, Westinghouse Plants.'' The Bases associated with this TS would also be revised to address the proposed change.

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

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

The current Salem Technical Specifications allows the use of administrative means to unisolate a containment isolation valve on an intermittent basis. The proposed change eliminates the potential for varying interpretations of the TS footnote by relocating it to the ACTION section of the Technical Specifications in accordance with the guidance of NUREG 1431, Rev 1 (April 1995) ''Standard Technical Specifications Westinghouse Plants (NUREG–1431).'' PSE&G [PSE&G] views the proposed change as a change that is editorial in nature.

The proposed change does not delete any existing surveillance requirements or delete any requirements from the Limiting Condition for Operations (LCOs) or Action Statements, and therefore does not reduce the actions that are currently taken in the TS to demonstrate operability of plant structures, systems, or components (SSCs). The proposed change continues to ensure the operability of the containment isolation valves, therefore ensuring that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment.

Since these changes do not modify any SSCs or reduce the current requirements for demonstrating operability of these SSCs, the proposed changes to the TS do not involve a significant increase in the probability or consequences of an accident previously evaluated in the Safety Analysis Report (SAR).

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

The proposed amendment eliminates the potential for varying interpretations of the TS footnote by relocating it to the ACTION section of the Technical Specifications in accordance with the guidance of NUREG 1431, Rev 1 (April 1995) ''Standard Technical Specifications Westinghouse Plants (NUREG–1431).''

The proposed change does not alter the physical configuration of the plant. The proposed change does not affect any systems, structures or components assumed to function in the accident analysis, or creates a new or different accident scenario. The proposed change to the TS does not affect the ability of the plant systems to meet their current TS requirements or design basis functions. Therefore, the proposed change does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR or create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendment eliminates the potential for varying interpretations of the TS footnote by relocating it to the ACTION section of the Technical Specifications in accordance with the guidance of NUREG 1431, Rev 1 (April 1995) ''Standard Technical Specifications Westinghouse Plants.'' The proposed amendment does not change any testing acceptance criteria or modify any protective trip setpoint. The proposed change will continue to ensure that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment.

There is no reduction in the current surveillance requirements required to demonstrate the operability of plant SSCs. 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, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Clifford.

Public Service Electric & Gas Company, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Units Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: April 13, 2000.

Description of amendment request: The proposed amendments would delete Technical Specification (TS) 3/4.1.3.2.2 which is related to shutdown and control rod group demand position indication in modes 3, 4, and 5.

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

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

The proposed TS change does not involve any physical changes to plant structures, systems or components (SSC). Shutdown margin will continue to be maintained as required by plant Technical Specifications to ensure the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. Shutdown and control rod group demand position indication is not required to ensure adequate shutdown margin in modes 3, 4 and 5 and therefore cannot contribute to the initiation of any accident. The proposed changes do not change or alter the design assumptions for the systems or components used to mitigate the consequences of an accident, and the initial conditions and methodologies used in the accident analyses remain unchanged. Therefore, accident analyses results are not impacted. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not involve any physical changes to plant structures, systems or components. The safety functions of the related structures, systems, or components are not changed in any manner, nor is the reliability of any structures, systems, or components reduced. No new or different type of equipment will be installed by this requested change. Therefore, no new failure modes or potential accident initiators are introduced. Therefore, the proposed amendments 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.

Shutdown margin will continue to be maintained in accordance with the requirements of TS 3/4.1.1. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. Therefore, the proposed amendments 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: Jeffrie J. Keenan, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: June 1, 2000.

Description of amendment request: The proposed amendments would revise the vessel pressure and temperature limit curves that are in the Technical Specifications.

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

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

The changes to the calculational methodology for the pressure and temperature (P/T) limits based upon Code Cases N–640 and N–588 continue to provide adequate margin in the prevention of a non-ductile type fracture of the reactor pressure vessel (RPV). The code cases were developed based upon the knowledge gained through years of industry experience. P/T curves developed using the allowances of Code Cases N–640 and N–588 indeed yield more operating margin. However, the experience gained in the areas of fracture toughness of materials and pre-existing undetected defects show that some of the existing assumptions used for the calculation of P/T limits are unnecessarily conservative and unrealistic. Therefore, providing the allowances of the subject code cases in developing the P/T limit curves will continue to provide adequate protection against nonductile-type fractures of the RPV.

The evaluation for extending the Unit 1 and Unit 2 P/T limit curves to 54 EFPYs was performed using the approved methodologies of 10 CFR 50, Appendix G, and with the allowances of code cases N–588 and N–640. The curves generated from these methods ensure the P/T limits will not be exceeded during any phase of reactor operation. Therefore, the probability of occurrence and the consequences of a previously analyzed event are not significantly increased. Finally, the proposed changes will not affect any other system or piece of equipment designed for the prevention or mitigation of previously analyzed events.

Thus, the probability of occurrence and the consequences of any previously analyzed event are not significantly increased as the result of the proposed changes.

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

The proposed changes provide more operating margin in the P/T limit curves for inservice leakage and hydrostatic pressure testing, non-nuclear heatup and cooldown, and criticality, with the benefits being primarily realizable during the pressure tests. The revised curves also extend the P/T limit curves to 54 EFPYs. However, operation in the ''new'' regions of the curves have been analyzed with the new P/T curves providing adequate protection against a nonductile-type fracture of the RPV. Otherwise, the proposed changes do not result in any new or unanalyzed operation of any system or piece of equipment important to safety, and as a result, the possibility of a new type event is not created.

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

As mentioned previously, the revised P/T curves provide more operating margin and thus, more operational flexibility than the current P/T curves. With the increased operational margin, a reduction in the safety margin results with respect to the existing curves. However, the industry experience since the inception of the P/T limits in 1974 confirms that some of the existing methodologies used to develop P/T curves are unrealistic and unnecessarily conservative. Accordingly, ASME Code Cases N–640 and N–588 take advantage of the acquired knowledge by establishing more realistic methodologies for the development of P/T curves. Therefore, operational flexibility is gained and an acceptable margin of safety to RPV non-ductile type fracture is maintained.

The extension of the P/T curves to 54 EFPYs was performed per the guidelines of 10 CFR 50, and using code cases N–640 and N–588 and thus, the margin of safety is not significantly reduced as the result of the proposed changes.

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

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

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: March 3, 2000.

Description of amendment request: The proposed amendments would revise technical specification (TS) 3.9.4, ''Containment Penetrations'', by allowing the equipment hatch to be open during core alterations and/or during movement of irradiated fuel within the containment.

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?

No. The proposed changes will allow the equipment hatch to be open during core alterations and movement of irradiated fuel assemblies inside containment. The existing [Vogtle Electric Generating Plant] VEGP TS allow the air lock doors to be open during core alterations and movement of irradiated fuel assemblies inside containment, and the dose analyses for a fuel handling accident inside containment remain bounding for the case of [an open equipment hatch]. The proposed changes will not alter the manner in which fuel is handled or core alterations are performed. Therefore the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed changes do not create any new failure modes for any system or component, nor do they adversely affect plant operation. No new equipment will be added and no new limiting single failures will be created. The plant will continue to be operated within the envelope of the existing safety analyses. Therefore, the proposed changes do not create the possibility of a new or different kind of accident previously evaluated.

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

No. The previously determined radiological dose consequences for a fuel handling accident inside containment with the air lock doors open remain bounding for the proposed changes. These previously determined dose consequences were determined to be well within the limits of 10 CFR 100 and they meet the acceptance criteria of [Standard Review Plan] SRP Section 15.7.4 and [General Design Criteria] GDC 19. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: May 17, 2000.

Brief description of amendments: The proposed amendment would change the Allowable Values specified in Technical Specification Table 3.3.5–1 to ensure that the 6.9 kilovolt (kV) and 480 volt (V) undervoltage relays initiate the necessary actions when required. In addition, some unnecessary limits would be deleted.

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

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

Response: No.

The proposed License Amendment Request includes more restrictive Allowable Values for the Preferred offsite source bus undervoltage function, the Alternate offsite source bus undervoltage function, the 6.9 kv Class 1E bus loss of voltage function, the 6.9 kv Class 1E bus degraded voltage function, and the 480 V Class 1E bus degraded voltage function. These more restrictive values assure that all applicable safety analysis limits are being met. The 480 V low grid undervoltage relay allowable value is being lowered to the same as the 480 V degraded voltage relays which matches its function. This is a less restrictive value but the value still assures that all applicable safety analysis limits are being met. Lowering of the 480 V low grid undervoltage allowable value will minimize unnecessary actuations that could challenge plant systems. Changing the 6.9 kV and 480 V degraded voltage, 480 V low grid undervoltage, the 6.9 kV loss of voltage, and the preferred and alternate bus undervoltage Allowable Values in the Technical Specifications has no impact on the probability of occurrence of any accident previously evaluated. Because all accident analyses continue to be met, these changes do not impact the consequences of any accident previously evaluated.

Removal of the upper limits for the preferred and alternate bus undervoltage and the lower limit for the 6.9 kV Class 1E bus loss of voltage relays does not impact the probability of occurrence of any accident previously evaluated. None of the accident analyses are affected, therefore, the consequences of all previously evaluated accidents remain unchanged.

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

Response: No.

None of the changes affect plant hardware or the operation of plant systems in a way that could initiate an accident. Therefore, the proposed 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 a margin of safety?

Response: No.

There were no changes made to any of the accident analyses or safety analysis limits as a result of this proposed change. Further, the proposed change does not affect the acceptance criteria for any analyzed event. Removal of the upper limits for the preferred and alternate source bus undervoltage and the lower limit for the 6.9 kV Class 1E bus loss of voltage relays does not change the margin of safety. Each allowable value, as revised, assures the safety analysis limits assumed in the safety analyses as discussed in Chapter 15 of the FSAR [Final Safety Analysis Report] is maintained. The margin of safety established by the Limiting Conditions for Operation also remains unchanged. Thus there is no effect on 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: George L. Edgar, Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

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

Date of application request: May 25, 2000 (ULNRC–04258).

Description of amendment request: The proposed amendment would expand (1) the range of acceptable lift settings for the pressurizer safety valves (PSVs), and (2) the tolerance (from +1% to +2%) of the as-found, measured lift settings of tested PSVs, to be operable. The as-left lift settings, following testing, of the PSVs would not be changed from the current range of +1%. The amendment would revise Technical Specifications (TS) 3.3.2, ''Engineered Safety Features Actuation System (ESFAS) Instrumentation,'' 3.4.10, ''Pressurizer Safety Valves,'' and 3.4.11, ''Pressurizer Power Operated Relief Valves (PORVs),'' of the Callaway TS. For TS 3.3.2, a new Action H for one or more trains inoperable would be added, the note for surveillance requirement (SR) 3.3.2.14 would be revised to identify another slave relay that the SR would be applicable to, and the automatic PORV actuation would be added to Table 3.3.2–1, ''Engineered Safety Features Actuation System Instrumentation.'' For TS 3.4.10, the range of allowable PSV lift settings in the limiting condition for operation (LCO) would be expanded from >2460 and <2510 to >2411 and <2509, and SR 3.4.10.1 would be revised to state that following testing, the lift settings shall be ''within 1% of 2460 psig'' instead of simply ''within 1%.'' The nominal PSV lift setting would be changed from 2485 psig to 2460 psig because the maximum PORV lift setting would not be increased and the minimum setting would be reduced 59 psig. For TS 3.4.11, Actions A and B would be revised to be actions for inoperable PORVs either solely due to excessive PORV seat leakage (Action A) or for reasons other than excessive seat leakage (Action B), and Action E would remain an action for two inoperable PORVs, but would be only for reasons other than excessive seat leakage. The licensee also provided corrections to the Bases of the TSs and the Callaway Final Safety Analysis Report (FSAR) for the above changes.

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

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

The pressurizer safety valves (PSVs), in conjunction with the Reactor Trip System (RTS), provide overpressure protection for the Reactor Coolant System (RCS). The PSV [lift] setpoint is established to maintain the RCS pressure below 110% of the system design pressure. The proposed change in the minimum allowable PSV setpoint could result in a transient being terminated at a pressure that is lower than that assumed in the transient's analysis. However, the primary system pressure boundary is not challenged by the minimum allowable PSV setpoint. Since the maximum allowable PSV setpoint is unaffected by the proposed change (other than from round-off, as discussed previously [in the application, from 2510 to 2509 psig]), the primary system pressure boundary is not challenged by the maximum allowable PSV setpoint.

With a nominal setpoint of 2460 psig and a [as-found] +2% setpoint tolerance, the PSV actuation setpoint could potentially open at pressures as low as 2410 psig (rounded up in revised LCO 3.4.10 to 2411 psig). This lower PSV actuation setpoint will reduce the margin between the pressurizer PORV and PSV actuation setpoint from 125 psi to 75 psi. A 75 psi margin is considered adequate and should not challenge the PSVs on Condition I transients.

The majority of the Callaway PRA [probabilistic risk assessment] event trees question the capability of the PORVs to open for RCS cooldown and depressurization or for feed and bleed cooling. Some event trees question the capability of the PORVs to reclose to terminate RCS depressurization and coolant inventory loss. The transient-induced ATWS [anticipated transient without scram] event trees question the capability of the PSVs to reclose after opening for these high pressure transients. The maximum allowable PSV setpoint is essentially unchanged; therefore, the proposed change will not adversely impact the probability of the PSVs failing open. Upgrading the automatic PORV actuation circuitry to fully Class 1E, and revising the Technical Specification operability and surveillance requirements to demonstrate the operability of the automatic PORV actuation circuitry, will enhance valve reliability and assure compliance with NRC Generic Letter 90–06. However, it has been determined that this plant modification increase the probability that the PORVs will inadvertently open and remain open if multiple transmitter failures are postulated. With the new safety grade PORV 2/4 [two out of four] opening actuation logic, two failed high pressurizer pressure channels would result in inadvertent opening of both PORVs and the PORVs would remain open until remote-manually closed. Since two of the four channels available to reclose the PORVs are assumed to have failed high, and since closure of the PORVs would require a 3/4 logic to close after the modification is implemented, there would be no signal to close the PORVs on a low pressurizer pressure signal. With the current opening logic, a single failed high pressurizer pressure channel would result in opening one PORV. However, the current 2/4 closure logic would reclose that PORV when pressurizer pressure drops below approximately 2200 psia. With the current control logic, three failed high pressurizer pressure channels (3/4) are required for both PORVs to inadvertently open and remain open. However, the consequences of both PORVs inadvertently opening and remaining open are bounded by the analysis in FSAR Section 15.6.1, ''Inadvertent Opening of a Pressurizer Safety or Relief Valve.'' Since a pressurizer safety valve is sized to relieve approximately twice the steam flow rate of a pressurizer PORV, and will therefore allow a much more rapid depressurization upon opening, the analysis in Section 15.6.1 examines the accidental depressurization of the RCS associated with an inadvertent opening of a pressurizer safety valve. While there is no way to isolate a stuck-open pressurizer safety valve, two open PORVs can be remote-manually isolated by either closing the PORVs or the PORV block valves. Since there is a small impact due to multiple channel failures resulting in an increase in the probability of both PORVs inadvertently opening and remaining open, it is concluded that the proposed activity increases the probability of occurrence of an accident previously evaluated in the FSAR. However, multiple failures are required for this malfunction and failure modes that result in multiple channels failing high are highly unlikely. Therefore, this increase in the probability that the PORVs will inadvertently open and remain open is considered to be insignificant.

All evaluations performed for overpressure transients conservatively assume the upper limit of the PSV tolerance as the pressure to which the RCS is subjected. It has been determined that the design transients are not adversely affected because the limiting transients are not sensitive to the pressure tolerance change. Although the lower PSV setpoint would result in a lower PSV relief flow rate, the slightly lower valve flow rate would be more than compensated for by the reduced valve opening pressure. The change to the PSV setpoint and setpoint tolerance does not change the conclusions of the existing thermal-hydraulic and stress analyses for the pressurizer safety and relief system. The design function of the valves is not being changed and the conclusions documented in the NRC Safety Evaluation of Callaway's response to NUREG–0737 Item II.D.1[''Performance Testing of the Pressurizer Power-Operated Relief Valve,''] (dated September 10, 1987) are unchanged (see also FSAR Section 18.2.5). The PORVs and associated discharge piping can accommodate water relief.

Overall protection system performance will remain within the assumptions of the previously performed accident analyses since the only hardware changes are associated with making the automatic PORV actuation circuitry fully Class 1E. The RTS and Engineered Safety Features Actuation System (ESFAS) protection systems will continue to function in a manner consistent with the plant design basis. The automatic PORV actuation circuitry modification will be performed in such a manner that all design, material, and construction standards that were applicable to safety-related systems prior to the change are maintained.

The proposed change will not affect the probability of any event initiators nor will the proposed change negatively affect the ability of any safety-related equipment to perform its intended function. Changing the PSV lift setting does not change the probability that an event will occur which will result in the PSV opening. There will be no degradation in the performance of safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters.

Since the FSAR Chapter 15 LOCA [loss-of-coolant accident], SGTR [steam generator tube rupture] and MSLB [main steam line break] analyses all result in decreasing RCS pressure and do not challenge the PSV opening pressure, none of these events are affected by the proposed change to the PSV nominal setpoint and the allowable setpoint tolerance. Timely operator actions will be taken to preclude water relief through the PSVs during an Inadvertent ECCS [emergency core cooling system] Actuation at Power event. Water relief from the PORVs for the latter event would result in a larger discharge of RCS inventory than currently analyzed, wherein operator action is assumed to terminate safety injection within 10 minutes prior to the pressurizer filling. However, FSAR Figure 15.5–3 in Attachment 5 [to the application] demonstrates that DNB [departure from nucleate boiling] is not a concern, there will be no fuel failures associated with this event, and RCS inventory will be directed to the pressurizer relief tank located inside containment. Therefore, there will be no impact on offsite radiological consequences. None of the other non-LOCA transients are adversely affected by the proposed change. Since none of the other FSAR Chapter 15 events are adversely affected, the radiological consequences of those events are not adversely affected.

In the Westinghouse reanalysis of the Inadvertent ECCS Actuation at Power event, the minimum PSV opening setpoint serves as a limit to demonstrate the acceptability of the assumed operator action times to assure that the PSVs will not be required to operate while the pressurizer is water solid. A lower PSV opening setpoint could potentially require earlier operator actions to prevent water relief through the PSVs. Simulator exercises for the Inadvertent ECCS Actuation at Power event were performed on the Callaway training simulator on August 10, 1999 to determine the times required for the control room operators to stop the NCP [normal charging pump] and unblock the PORVs and assure their availability for automatic pressure relief. In all cases, the NCP was stopped within four (4) minutes and the PORVs were unblocked and available for automatic pressure relief within seven (7) minutes. The reanalysis in Attachment 5 [to the application] conservatively credits operator actions from the main control room to stop the NCP in six (6) minutes and to unblock the PORVs and assure their availability for automatic pressure relief in nine (9) minutes. These times include all process and instrumentation delays. The revised FSAR Figure 15.5–2 shows that if operator actions are taken within these time frames to terminate NCP flow and to assure at least one PORV is available for automatic pressure relief, water relief through the PSVs is precluded. Procedure changes and periodic operator requalification training will provide assurance that these operator actions can be performed within the assumed time constraints.

Based on the above discussions, the proposed change will not involve a significant increase in the probability of occurrence or the consequences of an accidently previously evaluated.

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

The nominal setpoint for the PSVs will be lowered by 1% from 2485 psig to 2460 psig. The allowable setpoint tolerance will be increased from +1% to +2%. The combined effect of these changes results in a 2% decrease in the minimum acceptable PSV [lift] setpoint from 2460 psig to 2411 psig. The change in the PSV setpoint and in the tolerance of the setpoint does not change their ability to open on demand. The maximum acceptable PSV setpoint is unaffected by this proposed change, other than round-off as discussed previously. Since the FSAR accident analyses do not rely on the automatic actuation of non-safety related control grade systems or components for accident mitigation, a plant modification will make the automatic pressurizer PORV pressure relief circuitry fully Class 1E.

The proposed change to the PSV nominal setpoint and the allowable setpoint tolerance will not prevent the PSVs from performing their RCS overpressurization protection function. Additionally, the proposed change does not affect the ability of any other safety-related equipment to perform its safety function.

The only hardware changes are associated with making the automatic PORV actuation circuitry fully Class 1E. The RTS and Engineered Safety Feature Actuation System (ESFAS) protection systems will continue to function in a manner consistent with the plant design basis. The automatic PORV actuation circuitry modification will be performed in such a manner that all design, material, and construction standards that were applicable to safety-related systems prior to the change are maintained. While the possibility that the PORVs fail to control RCS pressure, that at least one PORV fails to open, and that the operator fails to open the block valve and assure the PORV(s) are available for automatic pressure relief within the required time frame are all malfunctions of a different type than currently analyzed in the FSAR, they do not create different accident types. The Class 1E upgrade and changes to Emergency Operating Procedure E–0 will provide assurance that the reanalysis presented in Attachment 5 [to the application] will bound the results of this event which, in turn, is also bounded by the results presented in FSAR Section 15.6.1 for an inadvertent PSV opening.

There are no other changes in the method by which any safety-related plant system performs its safety function. The change will not affect the normal method of plant operation.

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

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

The PSVs, in conjunction with the RTS, provide overpressure protection for the RCS. The change in the upper limit of the PSV tolerance from +1% to +2%, with a reduction in the nominal setpoint from 2485 psig to 2460 psig, does not challenge the upper limit of overpressure protection. The maximum opening pressure setpoint is unchanged (other than a conservative round-off), and therefore, does not impact analyses performed for overpressure transients. The change to the PSV setpoint and setpoint tolerance does not change the conclusions of the existing thermal-hydraulic and stress analyses for the pressurizer safety and relief system. For all non-LOCA events, the above evaluations support the change in the PSV setpoint and setpoint tolerance from 2485 psig +1% to 2460 psig +2%. The change in the PSV setpoint and setpoint tolerance also has no effect on the RTS and ESFAS trip setpoints.

The Bases for Technical Specification 3.4.10 states the following in the Background section:

''The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure * * * The relief capacity for each valve, 420,000 lb/hr at 2485 psig plus 3% accumulation, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer * * * .''

The locked RCP [reactor coolant pump] rotor and loss of external electrical load/turbine trip transient analyses assume PSV actuation at 2550 psia. This value is conservatively based on a nominal PSV setpoint of 2500 psia plus a 1% setpoint tolerance and a 1% setpoint shift (due to the presence of the water seal). The maximum allowable PSV setpoint of 2509 psig is unaffected by the proposed change, other than a conservative round-off discussed previously. At a pressure of 2509 psig, the minimum relief capacity of the safety valves would be in excess of 420,000 lb/hr. However, the safety analyses for overpressurization events conservatively assume a 420,000 lb/hr minimum design relief capacity for the PSVs.

The proposed change does not affect the acceptance criteria for any other analyzed event nor is there a change to any other Safety Analysis Limit (SAL). The acceptance criteria for the Inadvertent ECCS Actuation at Power event will remain the same as currently analyzed; however, operator action and automatic PORV actuation will be relief upon to demonstrate compliance with that event's acceptance criteria.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, DNBR limits, F_Q, $F\Delta H$, LOCA PCT [peak cladding temperature], peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the [NRC] Standard Review Plan continue to be met.

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

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

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

NRC Section Chief: Stephen Dembek.

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

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

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

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: June 1, 2000.

Description of amendment request: The proposed amendment would permit changes to the Perry Nuclear Power Plant Updated Safety Analysis Report (USAR) to incorporate descriptions (in the form of text, tables, and drawings) of modifications to the Emergency Service Water (ESW) alternate intake sluice gate. The modifications will include (1) installation of a safety-related Class 1E selector switch that will be used to disable the automatic opening function of the sluice gate during warm weather and (2) installation of a non-safety inflatable sealing device on the gates between the ESW forebay and the alternate intake tunnel. The modifications are designed to increase overall reliability of the ESW system and to eliminate undesired operation of the ESW pumps.

*Date of publication of individual notice in **Federal Register:*** June 14, 2000 (65 FR 37414).

Expiration date of individual notice: July 14, 2000.

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

NRC Section Chief: Anthony J. Mendiola.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, http://www.nrc.gov (the Electronic Reading Room).

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

Date of application for amendment: April 24, 2000.

Brief description of amendment: The amendment allowed a one-time extension of some Technical Specification surveillance intervals due to elimination of a planned midcycle outage. The surveillances would be extended to no later than November 30, 2000.

Date of issuance: June 12, 2000.

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

Amendment No.: 129.

Facility Operating License No. NPF–62: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 8, 2000 (65 FR 26642).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 12, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 15, 2000.

Brief description of amendments: The amendments revised the Technical Specifications to permit plant operation with an ultimate heat sink temperature of 100 °F.

Date of issuance: June 13, 2000.

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

Amendment Nos.: 107 and 107.

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

Date of initial notice in ***Federal Register:*** May 3, 2000 (65 FR 25763).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 13, 2000.

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: July 27, 1999, as supplemented by letters dated October 7, 1999, and May 31, 2000.

Brief description of amendments: The amendments revised the Technical Specifications by adding a surveillance requirement to verify the Keowee out-of-tolerance logic trips and blocks closure of the appropriate overhead or underground power path breakers.

Date of Issuance: June 6, 2000.

Effective date: As of the date of issuance and shall be implemented by November 30, 2000.

Amendment Nos.: 312, 312 and 312.

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

Date of initial notice in Federal Register: August 25, 1999 (64 FR 46429). The supplements dated October 7, 1999, and May 31, 2000, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Portland General Electric Company, et al., Docket No. 50–344, Trojan Nuclear Plant, Columbia County, Oregon

Date of application for amendment: November 16, 1999.

Brief description of amendment: The amendment revised the Permanently Defueled Technical Specifications by removing Figure 4.1–1, ''Site and Exclusion Area Boundaries,'' and incorporating the applicable portions of this figure in the Trojan Defueled Safety Analysis Report. Other associated administrative changes resulting from the deletion of Figure 4.1–1, as well as an administrative change to the table of contents, were also made.

Date of issuance: May 31, 2000.

Effective date: May 31, 2000.

Amendment No.: 204.

Facility Operating License No. NPF–1: The amendment changes the Permanently Defueled Technical Specifications.

Date of initial notice in Federal Register: January 26, 2000 (65 FR 4289).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 31, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 9, 2000.

Brief description of amendment: This amendment revises Technical Specification (TS) Limiting Condition for Operation 3.8.2.1 to add two new Action Statements for operating conditions where a Class 1E battery's electrolyte temperature is below the minimum limit specified in TS Surveillance Requirement 4.8.2.1.b.3.

Date of issuance: June 9, 2000.

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

Amendment No.: 127.

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

Date of initial notice in Federal Register: March 8, 2000 (65 FR 12294).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 28, 1999, as supplemented March 17, 2000.

Brief description of amendment: Revised Technical Specifications definitions for Engineered Safety Feature Response Time and Reactor Trip System Response Time, to provide for verification of response time for selected components, provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

Date of issuance: June 13, 2000.

Effective date: June 13, 2000.

Amendment No.: 24.

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

Date of initial notice in ***Federal Register:*** October 20, 1999 (64 FR 56534). The March 17, 2000, submittal provided clarifying information that did not change the scope of the original request or change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 13, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: November 8, 1999, as supplemented by letters dated April 13, and May 30, 2000.

Brief description of amendments: The amendments change Technical Specification 5.5.11, ''Ventilation Filter Testing Program (VFTP),'' to include the requirement for laboratory testing of Engineered Safety Feature (ESF) Ventilation System charcoal samples per American Society for Testing and Materials D3803–1989 and the application of a safety factor of 2.0 to the charcoal filter efficiency assumed in the plant design-basis dose analyses. The license amendments also extend the implementation date for License Amendment 74, currently June 30, 2000, to December 31, 2000.

Date of issuance: June 12, 2000.

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

Amendment Nos.: 78 and 78.

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

Date of initial notice in ***Federal Register:*** December 29, 1999 (64 FR 73101). The April 13, and May 30, 2000, letters provided clarifying information that did not change the scope of the November 8, 1999, 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 June 12, 2000.

No significant hazards consideration comments received: No.

Viacom Inc., Docket No. 50–22, Test Reactor, Waltz Mill, Pennsylvania

Date of application for amendment: February 14, 2000 supplemented on March 8 and 25, 2000.

Brief description of amendment: This amendment changes the license to reflect the transfer of the licensee for the Test Reactor at Waltz Mill from the CBS Corporation to Viacom Inc.

Date of issuance: May 31, 2000.

Effective Date: May 4, 2000.

Amendment No.: 12.

Facility License No. TR–2: This amendment changes the license.

Date of Initial notice in ***Federal Register:*** February 29, 2000 (65 FR 10841).

The Commission has issued a Safety Evaluation for this amendment dated April 13, 2000.

No significant hazards consideration comments received: No.

Local Public Document: N/A.

Dated at Rockville, Maryland, this 21st day of June 2000.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–16193 Filed 6–27–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

State of Oklahoma: NRC Staff Assessment of a Proposed Agreement Between the Nuclear Regulatory Commission and the State of Oklahoma

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of a proposed agreement with the State of Oklahoma.

SUMMARY: This notice is announcing that the Nuclear Regulatory Commission (NRC) has received a request from Governor Frank Keating of Oklahoma that the NRC consider entering into an Agreement with the State as authorized by Section 274 of the Atomic Energy Act of 1954, as amended (Act). Section 274 of the Act contains provisions for the Commission to enter into agreements with the Governor of any State providing for the discontinuance of the regulatory authority of the Commission. Under the proposed Agreement, submitted December 28, 1999, the Commission would discontinue and Oklahoma would take over portions of the Commission's regulatory authority over radioactive material covered under the Act within the State of Oklahoma. In accordance with 10 CFR 150.10, persons, who possess or use certain radioactive materials in Oklahoma, would be released (exempted) from portions of the Commission's regulatory authority under the proposed Agreement. The Act requires that NRC publish those exemptions. Notice is hereby given that the pertinent exemptions have been previously published in the **Federal Register** and are codified in the Commission's regulations as 10 CFR Part 150. NRC is publishing the proposed Agreement for public comment, as required by the Act. NRC is also publishing the summary of an assessment conducted by the NRC staff of the proposed Oklahoma byproduct material regulatory program. Comments are invited on (a) the proposed Agreement, especially its effect on public health and safety, and (b) the NRC staff assessment.

DATES: The comment period expires July 7, 2000. Comments received after this date will be considered if it is practical to do so, but the Commission cannot assure consideration of comments received after the expiration date.

ADDRESSES: Written comments may be submitted to Mr. David L. Meyer, Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, Washington, DC 20555–0001. Copies of comments received by NRC may be examined at the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC. Copies of the proposed Agreement, copies of the request for an Agreement by the Governor of Oklahoma including all information and documentation submitted in support of the request, and copies of the full text of the NRC staff assessment are also available for public inspection in the NRC's Public Document Room.

FOR FURTHER INFORMATION CONTACT: Patricia M. Larkins, Office of State and Tribal Programs, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Telephone (301) 415–2309 or e-mail *pml@nrc.gov.*

SUPPLEMENTARY INFORMATION: Since Section 274 of the Act was added in 1959, the Commission has entered into Agreements with 31 States. The Agreement States currently regulate approximately 16,000 agreement material licenses, while NRC regulates approximately 5800 licenses. Under the proposed Agreement, approximately 220 NRC licenses will transfer to Oklahoma. NRC periodically reviews the performance of the Agreement States to assure compliance with the provisions of Section 274. Section 274e requires that the terms of the proposed Agreement be published in the **Federal Register** for public comment once each week for four consecutive weeks. This notice is being published in fulfillment of the requirement.

I. Background

(a) Section 274d of the Act provides the mechanism for a State to assume regulatory authority, from the NRC, over certain radioactive materials [1] and activities that involve use of the materials. In a letter dated December 28, 1999, Governor Keating certified that the State of Oklahoma has a program for the control of radiation hazards that is adequate to protect public health and safety within Oklahoma for the materials and activities specified in the proposed Agreement, and that the State desires to assume regulatory responsibility for these materials and activities. Included with the letter was the text of the proposed Agreement, which is included as Appendix A to this notice.

The radioactive material and activities (which together are usually referred to as the ''categories of material'') which the State of Oklahoma requests authority over are: (1) The possession and use of byproduct materials as defined in Section 11e.(1) of the Act; (2) the possession and use of special nuclear material in quantities not sufficient to form a critical mass; (3) the regulation of the land disposal of byproduct source or special nuclear material received from other persons; and (4) source material used to take advantage of its density and high mass properties where the use of the specifically licensed source material is subordinate to the primary specifically licensed use of either 11e.(1) byproduct material or special nuclear material, as provided for in regulations or orders of the Commission.

(b) The proposed Agreement contains articles that:

—Specify the materials and activities over which authority is transferred;
—Specify the activities over which the Commission will retain regulatory authority;
—Continue the authority of the Commission to safeguard nuclear materials and restricted data;

[1] The radioactive materials, sometimes referred to as agreement materials, are: (a) Byproduct materials as defined in Section 11e.(1) of the Act; (b) byproduct materials as defined in Section 11e.(2) of the Act; (c) source materials as defined in Section 11z. of the Act; and (d) special nuclear materials as defined in Section 11a. of the Act, restricted to quantities not sufficient to form a critical mass.