with Section 3303(a)(1), FUTA.[2] However, because the Department of Labor strongly supports endeavors to employ public assistance recipients, the Department is exploring legally permissible avenues that might benefit employers who hire welfare recipients. We will notify States of the findings upon completing the effort.

5. *Action.* State administrators are requested to take necessary action to assure that State law conforms with and is applied consistently with Section 3303(a)(1), FUTA, as interpreted in this UIPL.

6. *Inquiries.* Please direct inquiries to the appropriate Regional Office.

[FR Doc. 99–3266 Filed 2–9–99; 8:45 am]

BILLING CODE 4510–30–P

[2] Although the Department of Labor has not developed a comprehensive noncharging policy, noncharging based on pre-employment income or circumstances is prohibited, because, as explained above, it is plainly inconsistent with Federal law.

DEPARTMENT OF LABOR

Employment and Training Administration

Job Training Partnership Act: Native American Employment and Training Council

AGENCY: Employment and Training Administration, Labor.

ACTION: Notice of meeting.

SUMMARY: Pursuant to section 10(a)(2) of the Federal Advisory Committee Act (Pub. L. 92–463; 5 U.S.C. App. Sec. 10), as amended, and section 401(k)(1) of the Job Training Partnership Act, as amended [29 U.S.C. 1671(k)(1)], notice is hereby given of a meeting of the Native American Employment and Training Council.

TIME AND DATE: The meeting will begin at 9:00 a.m. EST on Thursday, February 25, 1999, and continue until 5:00 p.m. EST that day. The meeting will reconvene at 9:00 a.m. EST on Friday, February 26, 1999, and adjourn at 4:00 p.m. EST on that day. The period from 3:00 p.m. to 5:00 p.m. EST on February 25 will be reserved for participation and presentation by members of the public.

PLACE: On Thursday, February 25, Room S–1011, and on Friday, February 26, Rooms N–5437 A, B, and C of the Frances Perkins Building, U.S. Department of Labor, 200 Constitution Avenue, N.W., Washington, D.C. 20210.

STATUS: The meeting will be open to the public.

MATTERS TO BE CONSIDERED: The agenda will focus on the following topics: (1) status of the Program Year 1998 Partnership Plan; (2) progress of the evaluation of the section 401 program; (3) progress of the performance measures/standards workgroup; (4) status of technical assistance and training provision for Program Year 1998 and 1999; (5) status of FY 1999 Indian and Native American Welfare-to-Work program implementation; and (6) status of pending implementation of the Workforce Investment Act, including a report on the progress of the Regulations Work Group.

FOR FURTHER INFORMATION CONTACT: Ms. Anna W. Goddard, Director, Office of National Programs, Employment and Training Administration, U.S. Department of Labor, Room N–4641, 200 Constitution Avenue, NW, Washington, DC 20210. Telephone: (202) 219–5500, ext 122 (VOICE), or (202) 326–2577 (TDD) (these are not toll-free numbers).

Signed at Washington, DC, this 4th day of February, 1999.

Anna W. Goddard,

Director, Office of National Programs.

[FR Doc. 99–3267 Filed 2–9–99; 8:45 am]

BILLING CODE 4510–30–M

NATIONAL SCIENCE FOUNDATION

Sunshine Act Meeting

AGENCY HOLDING MEETING: National Science Foundation, National Science Board.

DATE AND TIME: February 17, 1999, 9:00 a.m.—Open Session.

PLACE: The G. Paul Getty Trust, 1200 Getty Center Drive, Getty Research Institute Lecture Hall, Los Angeles, CA 90049–1681.

STATUS: This meeting will be open to the public.

MATTERS TO BE CONSIDERED:

Wednesday, February 17, 1999

Open Session (9:00 a.m.–12:00 noon)

—Chairman's Report
—Director's Report
—Framework for Revising the NSF Strategic Plan
—Presentation: Demographic Considerations in Human Resources Development
—NSB Report on Achievement in Science and Mathematics Education
—Other Business

Open Session (2:30 p.m.–6:00 p.m.)

—Welcoming Remarks and Keynote Address
—Symposium on Environmental Research, Education and Assessment
—Session 1: Emerging Interdisciplinary Opportunities

Thursday, February 18, 1999

Open Session (8:30 a.m.–12:00 noon)

—Symposium on Environmental Research, Education and Assessment, continued
Session 2: New tools, Connections, Ways of Thinking
Session 3: Ethics and Equity

Open Session (1:30 p.m.–5:00 p.m.)

—Symposium on Environmental Research, Education and Assessment, continued
Session 4: From Reaction to Proaction
Session 5: Enabling Partnerships

Marta Cehelsky,

Executive Officer.

[FR Doc. 99–3408 Filed 2–8–99; 2:23 pm]

BILLING CODE 7555–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from January 15, 1999, through January 29, 1999. The last biweekly notice was published on January 27, 1999 (64 FR 4152).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administration 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 March 12, 1999, 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 at the local public document room for the particular facility involved. 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 aspects(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 preset 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:* Rulemakings and Adjudications Staff, 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 at the local public document room for the particular facility involved.

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

Date of amendment request: December 7, 1998.

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) to permit a one-time only extension of the steam generator tube inspection interval for fuel cycle 14 and delete the requirement to have NRC staff concurrence of the steam generator examination program. Specifically, TS 4.13A.2.a would be revised with a footnote that states ''Examinations scheduled for 1999 only, shall be conducted during the 2000 Refueling Outage which will commence no later than June 3, 2000. The scheduled examinations will be completed prior to return to service from the 2000 Refueling Outage.'' In addition, TS 4.13C.1 would be revised to state ''The proposed steam generator examination program shall be submitted for NRC staff review at least 60 days prior to each scheduled examination.''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 59.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

The proposed change does not involve any physical modifications to the plant or modification in the methods of plant operation which could increase the probability or consequences of previously evaluated accidents. The proposed change permits an extension of the current steam generator tube inservice inspection cycle. This extension would allow the steam generator tube examinations to be conducted during the 2000 refueling outage which will commence no later than June 3, 2000. The basis for acceptance of this increase in the technical specification limit is the ''non-operating'' steam generator time between the last examination and the upcoming examination. Extending the steam generator ''operating'' duration by 48 days would not significantly increase wear which might lead to tube failure. No appreciable steam generator tube wear or degradation is expected as a result of this extension. This change will not affect the scope, methodology, acceptance limits and corrective measures of the existing steam generator tube examination program. The probability and consequences of failure of the steam generators due to leaking or degraded tubes is not increased by the proposed change. Additionally the proposed administrative change to delete the requirement to receive NRC concurrence of the proposed steam generator examinations will have no bearing on the actual results of the steam generator examinations. Therefore, the probability and the consequence of a design basis accident are not being increased by the proposed change.

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

Plant systems and components will not be operated in a different manner as a result of the proposed Technical Specification change. The proposed change permits the upcoming steam generator tube examination to be conducted during the 2000 refueling outage that will commence no later than June 3, 2000. There are no plant modifications or changes in methods of operation. This extension is based upon the ''non-operating'' steam generator time between the last examination and the upcoming examination. Extending the steam generator ''operating'' duration by an additional 48 days would not significantly increase wear which might lead to tube failure. The proposed extension will not increase the probability of occurrence of a tube rupture, increase the probability or consequences of an accident, or create any new accident precursor. Additionally the proposed administrative change to delete the requirement to receive NRC concurrence of the proposed steam generator examinations will have no bearing on the actual results of the steam generator examinations. Therefore, the possibility of an accident of a different type than was previously evaluated in the safety analysis report is not created by the proposed change to the Technical Specification.

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

The proposed change to Technical specification section 4.13A.2.a will not reduce the margin of safety. This amendment involves an extension of the current steam generator tube inservice inspection cycle. The basis for acceptance of this increase in the technical specification limit is the ''non-operating'' steam generator time between the last examination and the upcoming examination. Extending the steam generator ''operating'' duration by an additional 48 days would not significantly increase wear which might lead to tube failure. No appreciable steam generator tube wear or degradation is expected as a result of this extension. Additionally the proposed administrative change to delete the requirement to receive NRC concurrence of the proposed steam generator examinations will have no bearing on the actual results of the steam generator examinations. Therefore, the accident analysis assumptions for design basis accidents are unaffected and the margin of safety is not decreased by the proposed Technical Specification change.

[* * *]

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.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

Attorney for licensee: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Project Director: S. Singh Bajwa, Director.

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

Date of amendment request: April 30, 1998.

Description of amendment request: The proposed amendment revises the definition of quadrant power tilt to clearly allow the use of either the incore detectors or the excore detectors for determining quadrant power tilt.

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

Criterion 1—Does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to the quadrant power tilt (QPT) definition will not alter any Safety Analysis Report (SAR) assumptions established and implemented by the technical specifications. The proposed change will allow the use of either the incore detectors or the excore power range detectors for determining QPT. This change is consistent with the improved Standard Technical Specifications (STS) which has been previously approved by the NRC. QPT measured by incore detectors provides a more accurate indication of reactor core power distribution than the value determined from the excore detectors. The accident prevention and mitigation features of the plant are not affected by this proposed amendment.

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

Criterion 2—Does not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed change to the definition of QPT does not alter the ANO–1 SAR analysis or core operating limits report (COLR). The change will clearly permit the use of either the incore detectors or the excore detectors for monitoring QPT. The design and physical configuration of the plant are not affected by this change.

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

Criterion 3—Does not involve a significant reduction in the margin of safety.

The proposed change to the QPT definition incorporates the improved TS definition contained in NUREG–1430. The revised definition allows the use of either the incore detectors or the excore power range detectors for determination of QPT. The change does not vary or affect any of the plant's operating parameters. The COLR currently specifies acceptable QPT limits based upon the measurement techniques. These limits are based upon the unique measurement characteristics of the incore and excore power range detectors and assure the measurement independent limit is not violated.

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

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

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

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

NRC Project Director: John N. Hannon.

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

Date of amendment request: August 6, 1998.

Description of amendment request: The proposed amendment revises the minimum and the maximum concentration limits for the sodium hydroxide tank. The proposed change also revises the minimum specified tank volume to refer to the parameter used in the analysis with no allowance for instrument uncertainty and deletes the maximum specified tank volume.

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

Criterion 1—Does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Sodium hydroxide is not an accident initiator. It is, however, a contributor to the mitigation of the effects of a Loss-of-Coolant-Accident (LOCA). The proposed change in NaOH tank concentration results in changing the expected post-LOCA reactor building sump pH. The reduction in the lower value of sump pH, from 8.5 to 7.0, is acceptable based on guidance contained in NUREG–0800, Standard Review Plan, Section 6.5.2, ''Containment Spray as a Fission Product Cleanup System Review Responsibilities,'' Revision 2, December 1988. This guidance allows the assumption of long-term iodine retention when the equilibrium sump pH, after mixing and dilution with the primary coolant and ECCS injection, is above 7.0. Although the change allows the volume of the NaOH tank to be maintained at a lower volume, the proposed minimum volume bounds the analyses of concern.

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

Criterion 2—Does not create the possibility of a new or different kind of accident from any previously evaluated.

Sodum hydroxide is added for iodine removal and for pH adjustment of the borated water in the reactor building sump following a LOCA. The proposed changes in NaOH tank concentration and volume introduce no new mode of plant operation.

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

Criterion 3—Does not involve a significant reduction in the margin of safety.

The proposed change in NaOH tank concentration results in changing the expected post-LOCA reactor building sump pH. This proposed change does involve an incremental reduction in the margin to safety since iodine retention is dependent on the pH of the sump/spray solution. However, this reduction is not considered significant in that the effect of the change in sump pH, from 8.5 to 7.0 has a relatively minor effect on iodine retention, as supported by Standard Review Plan (NUREG–0800), Section 6.5.2, Revision 2, dated December 1988. Although the change allows the volume of the NaOH tank to be maintained at a lower volume, the proposed minimum volume bounds the analyses of concern.

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

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

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

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

NRC Project Director: John N. Hannon.

Entergy Operations, Inc., Et Al., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi and Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: January 12, 1999, superceding the amendment request in the letter of September 30, 1996, for both stations.

Description of amendment request: The proposed amendment would add an additional required action to the Limiting Condition for Operation (LCO) 3.9.1, ''Refueling Equipment Interlocks,'' of the Technical Specifications for both stations. The additional action would allow an alternative to the current action for one or more inoperable refueling equipment interlocks. The current action is to ''suspend in-vessel fuel movement with equipment associated with the inoperable interlock(s).'' The alternative action proposed is to (1) insert a control rod withdrawal block, and (2) verify all control rods are fully inserted in core cells containing one or more fuel assemblies. The proposed amendment would also revise the Bases for the LCO 3.9.1 actions to describe the proposed alternative actions. The previous **Federal Register** notice of the amendment request in the superceded letter of September 30, 1996, was issued on June 16, 1996, (61 FR 31178), for Grand Gulf Nuclear Station (GGNS).

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:

I. The proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

The refueling interlocks are explicitly assumed in the GGNS Updated Final Safety Analyses Report (UFSAR) and RBS Updated Safety Analyses Report (USAR) analysis of the control rod removal error or fuel loading error during refueling. This analysis evaluates the probability and consequences of control rod withdrawal during refueling. Criticality and, therefore, subsequent prompt reactivity excursions are prevented during the insertion of fuel, provided all required control rods are fully inserted during the fuel insertion. The refueling interlocks accomplish this by preventing loading fuel into the core with any control rod withdrawn, or by preventing withdrawal of a rod from the core during fuel loading.

When the refueling interlocks are inoperable the current method of preventing the insertion of fuel when a control rod is withdrawn is to prevent fuel movement. This method is currently required by the Technical Specifications. An alternate method to ensure that fuel is not loaded into a cell with the control rod withdrawn is to prevent control rods from being withdrawn and verify that all control rods required to be inserted are fully inserted. The proposed actions will require that a control rod block be placed in effect thereby ensuring that control rods are not subsequently inappropriately withdrawn. Additionally, following placing the control rod withdrawal block in effect, the proposed actions will require that all required control rods be verified to be fully inserted. This verification is in addition to the requirements to periodically verify control rod position by other Technical Specification requirements. These proposed actions will ensure that control rods are not withdrawn and cannot be inappropriately withdrawn because an electrical or hydraulic block to control rod withdrawal is in place. Like the current requirements the proposed actions will ensure that unacceptable operations are blocked (e.g., loading fuel into a cell with a control rod withdrawn except following the requirements of LCO 3.10.6, ''Multiple Control Rod Removal—Refueling,'' which is unaffected by this change).

The proposed additional acceptable Required Actions provide an equivalent level of assurance that fuel will not be loaded into a core cell with a control rod withdrawn as the current Required Action or the Technical Specification Surveillance Requirement. Therefore, the proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

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

The change in the Technical Specification requirements does not involve a change in plant design. The proposed requirements will continue to ensure that fuel is not loaded into the core when a control rod is withdrawn except following the requirements of LCO 3.10.6, ''Multiple Control Rod Removal-Refueling,'' which is unaffected by this change.

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

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

As discussed in the Bases for the affected Technical Specification requirements, inadvertent criticality is prevented during the insertion of fuel provided all required control rods are fully inserted during the fuel insertion. The refueling interlocks function to support the refueling procedures by preventing control rod withdrawal during fuel movement and the inadvertent loading of fuel when a control rod is withdrawn.

The proposed change will allow the refueling interlocks to be inoperable and fuel movement to continue only if a control rod withdrawal block is in effect and all required control rods are verified to be fully inserted. These proposed Required Actions provide an equivalent level of protection as the refueling interlocks by preventing a configuration which could lead to an inadvertent criticality event. The refueling procedures will continue to be supported by the proposed required actions because control rods cannot be withdrawn and as a result fuel cannot be inadvertently loaded when a control rod is withdrawn except following the requirements of LCO 3.10.6, ''Multiple Control Rod Removal—Refueling,'' which is unaffected by this change.

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

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

Local Public Document Room Location: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120, for Grand Gulf Nuclear Station, and Government Documents Department, Louisiana State University, Baton Rouge, LA 70803, for River Bend Station.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502, for Grand Gulf Nuclear Station, and Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005, for River Bend Station.

NRC Project Director: John N. Hannon.

Florida Power and Light Company, Et Al., Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of amendment request: November 22, 1998.

Description of amendment request: The proposed amendment would revise the reactor thermal margin safety limit lines and flow rates stated in the technical specifications (TS). The amendment would also update the reference for dose conversion factors used in Dose Equivalent Iodine-131 calculations, and administrative changes to the criticality analysis uncertainty described in TS 5.6.1.a.1, update the analytical methods used in determining core operating limits listed in TS 6.9.1.11, and revise the TS bases for the steam generator pressure-low trip setpoint.

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

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

Replacement of the St. Lucie Unit 1 steam generators in 1997 resulted in an increase in RCS [reactor coolant system] flow. The proposed amendment would increase the values of design minimum reactor coolant flow and the low flow trip setpoint presently stated in the Technical Specifications (TS). These revisions are accompanied by a corresponding change to the Thermal Margin Safety Limit Lines of TS Figure 2.1–1. The RCS flow related revisions do not change the probability of any previously evaluated accident, as they do not impact any plant component, structure or system affecting the accident initiators. The proposed changes would continue to maintain adequate operational margin to TS limits for RCS flow and the low-flow trip setpoint.

The proposed changes to the thyroid dose conversion factors from TID–14844 to ICRP–30, fuel storage TS 5.6.1.a.1, the list of analytical methods in TS 6.9.1.11, and the Bases for Steam Generator Pressure-Low trip setting have no relevance to the accident initiators, and thus do not affect the frequency of occurrence of previously analyzed transients. Additionally, there are no changes to any active plant component due to these proposed changes.

The supporting evaluation of proposed TS changes demonstrates acceptable results for all the accidents previously analyzed, and it is concluded that the radiological consequences would remain within their established acceptance criteria when including the effects of increased RCS flow, increased low flow trip setpoint, and change to the thyroid dose conversion factors used in the determination of dose consequences. Proposed changes to the Bases for the Steam Generator Pressure-Low trip setpoint, fuel storage design features, and the list of analytical methods in TS 6.9.1.11 are administrative in nature and do not impact current safety analyses.

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

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

This proposed amendment revises limiting flow parameters to derive analysis benefits from increased RCS flow due to the replacement stream generators, while assuring safe plant operation commensurate with the proposed RCS flow and low flow trip setpoint changes. These changes along with the proposed changes to the Bases for the Steam Generator Pressure-Low trip setpoint, dose conversion factors, the list of analytical methods in TS 6.9.1.11, and the fuel storage design features do not require modifications to the plant configuration, systems or components which would create new failure modes. There would be no change in the modes of operation of the plant. The design functions of all the safety systems remain unchanged. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The proposed amendment revises limiting flow parameters to derive analysis benefits from increased RCS flow due to the replacement steam generators, while assuring safe plant operation commensurate with the proposed design minimum RCS flow and low-flow trip setpoint changes. FPL has evaluated the impact of the proposed changes on available margin to the acceptance criteria for Specified Acceptable Fuel Design Limits (SAFDL), 10 CFR 50.46(b) requirements, primary and secondary over-pressurization, peak containment pressure, potential radioactive releases, and existing limiting conditions for operation. With the proposed changes to the design minimum RCS flow, low-flow trip setpoint, and dose conversion factors, FPL has concluded that there would be no adverse impact to the existing safety analyses. The proposed changes to the Bases for the Steam Generator Pressure-Low trip setpoint, the list of analytical methods in TS 6.9.1.11, and the fuel storage design features are administrative in nature. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

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

NRC Project Director: Cecil O. Thomas.

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

Date of amendment request: December 18, 1998.

Description of amendment request: The proposed amendment would revise the St. Lucie Unit 2 Plant Technical Specifications (TS) Index Page III; TS 1.10, Dose Equivalent I–131; TS 2.1.1.2, Linear Heat Rate; Bases 2.1.1, Reactor Core; Bases Figure B2.1–1, Axial Power Distributions for Thermal Margin Safety Limits; Bases 2.2.1, Reactor Trip Setpoints (Variable Power Level-High); TS 3.1.1.1/4.1.1.1.1, Shutdown Margin—Tavg Greater Than 200 °F; TS 3/4.1.1.2, Shutdown Margin—Tavg Less Than or Equal to 200 °F; TS 3.1.2.2, Boration Systems Flow Paths—Operating; TS 3.1.2.4, Charging Pumps—Operating; TS 3.1.2.6, Boric Acid Makeup Pumps—Operating; TS 3.1.2.8, Borated Water Sources—Operating; Bases 3/4.1.1.1 and 3/4.1.1.2, Shutdown Margin; Bases 3/4.1.2, Boration Systems; and TS 6.9.1.11, Core Operating Limits Report (COLR). The core operating limits for shutdown margin will be relocated to the St. Lucie Unit 2 COLR.

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

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment involves changes to the dose conversion factors used in the thyroid dose calculations and the relocation of the SHUTDOWN MARGIN requirements for Modes 1 through 5 from TS to the Core Operating Limits Report (COLR). Additionally, the peak linear heat rate value corresponding to centerline melt is deleted from the TS. The deletion of this TS remains consistent with the requirements of 10 CFR 50.36. Bases Figure B2.1–1 is replaced with a new figure, consistent with the input assumptions of the safety analysis report.

The proposed amendment addresses analytical methods changes such as the use of HERMIT code in one dimensional mode for spatial details, the rod bow penalty calculations using L^2/I dependence discussed in CEN–289 (A)–P, CEAW methodology change for crediting the delta-T power trip, and the methodology for core designs containing Gadolinia-Urania burnable absorbers (CENPD–275–P, Revision 1–P, Supplement 1–P). None of these changes is a contributor to the initiation of previously evaluated accidents. The changes to TS bases and the COLR methodology changes have no impact on the accident initiators. Accordingly, the probability of an accident previously evaluated is not significantly increased.

The proposed changes have been evaluated by Florida Power & Light (FPL) and Asea Brown Boveri—Combustion Engineering (ABB–CE). The safety analyses assumed bounding physics parameters, and satisfy all the applicable acceptance criteria. Although specification 2.1.1.2 is deleted from TS, the safety analyses continue to meet the same centerline melt acceptance criteria as before and from which the peak linear heat rate value is derived. Additionally, the peak linear heat rate value (corresponding to the centerline melt) does not meet the criteria specified in 10 CFR 50.36 for safety limits.

The changes to TS bases do not affect safety analysis results. The relocation of SHUTDOWN MARGIN requirements to COLR does not affect analysis results or consequences as the limits remain unchanged. Future changes to these limits will be controlled per Generic Letter 88–16 under the provisions of 10 CFR 50.59.

The use of HERMITE code in one dimension, for space-time loss-of-flow simulation, has been successfully applied for other ABB–CE plants. The use of HERMITE code in this mode, for St. Lucie Unit 2, is acceptable since there are no fundamental core and nuclear steam supply system (NSSS) differences between St. Lucie Unit 2 and these plants. The analyses presented in this submittal include the use of a supplement to the gadolinia-urania core design methodology topical report. The change in the rod bow penalty effects similar to that approved for another ABB–CE plant is justified for St. Lucie Unit 2 based on a comparative analysis of factors influencing the rod bow. The change in the CEAW analysis method removes unnecessary conservatisms as compared to the previous analysis method. The validity of results and conclusions of this evaluation are contingent upon NRC approval of these revised methods.

The radiological does consequences for applicable safety analyses, using the dose conversion factors from ICRP–30, Supplement to Part 1, satisfy the acceptance criteria established to ensure compliance with the 10 CFR 100 dose limits.

The COLR methodology changes proposed to be listed in TS are those previously approved for CE plants with changes as described above. The use of these methodologies remains consistent with their applicability for safety analyses.

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

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

The proposed amendment involves changes to the Technical Specifications for the dose conversion factors used in the thyroid dose calculations, the deletion of TS 2.1.1.2, the replacement of Bases Figure B2.1–1, and the relocation of SHUTDOWN MARGIN requirements to the COLR. Additionally, there are methodology changes related to the safety analyses reported in this submittal. The methodology changes include the use of HERMITE code in one dimensional mode for space-time loss-of-flow simulations, revised rod bow DNB penalty calculations, CEAW analysis methodology change including the use of delta-T power trip, and a supplement to the methodology for core designs containing Gadolinia-Urania burnable absorbers (CENPD–275–P Revision I–P, Supplement I–P). None of these changes, including those of the TS bases, will affect the plant configuration and there will be no impact on any system performance.

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

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

The proposed changes to the Technical Specifications have been evaluated with respect to the safety analyses using either previously approved methodology or methodology currently under NRC review (CENPD–275–P, Revision I–P, Supplement I–P). The use of HERMITE code in one-dimensional mode for spatial details, for space-time loss-of-flow simulation, provides more accurate data for thermal margin calculations and has been used for similar applications at other plants. The calculations of rod bow DNB penalty using L^2/I dependence has been previously approved for another ABB–CE plant and is justified for St. Lucie Unit 2 based on an analysis of important factors influencing the rod bow. The CEAW methodology change showed acceptable analysis results after conservatively accounting for appropriate uncertainties.

The safety analyses performed with this methodology used bounding physics parameters to allow flexibility for future cycles core designs. The revised Bases Figure B2.1–1 is consistent with the attached safety analysis report. Deleting TS 2.1.1.2 is justified since the specified limit does not meet any of the criteria of 10 CFR 50.36, and the fuel centerline melt criteria applied to the Specified Acceptable Fuel Design Limit (SAFDL) is not changed. The setpoint analyses and safety analyses of all design basis accidents meet the applicable acceptance criteria with respect to the radiological consequences, SAFDLs, primary and secondary overpressurization, and 10 CFR 50.46 requirements. The proposed amendment, therefore, will not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

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

NRC Project Director: Cecil O. Thomas.

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

Date of amendment request: December 16, 1998.

Description of amendment request: The proposed amendment would revise Technical Specification 6.3, ''Unit Staff Qualifications,'' and add specific staff qualifications for a Multi-Discipline Supervisor position.

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

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed changes are administrative in nature addressing personnel qualification issues. The Multi-Discipline Supervisor (MDS) position will be filled with personnel who are experienced in one or more technical disciplines (maintenance, operations, engineering, or other related technical discipline). Fundamental working knowledge of tasks being performed will be acquired through the MDS initial training program. The training concentrates on developing the skills and knowledge of an MDS to safely oversee tasks for multi-discipline work teams. Therefore, four years experience in any related technical discipline or disciplines combined with the MDS training program provide adequate technical knowledge for proper job oversight. These proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated because they do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. Therefore, operation of either facility in accordance with its proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The changes being proposed are administrative in nature and do not affect assumptions contained in plant safety analyses the physical design and/or modes of plant operation defined in the facility operating license, or Technical Specifications that preserve safety analysis assumptions. These changes address qualification requirements for the MDS position. Since the proposed changes do not change the qualifications for those individuals responsible for the actual licensed operation of the facility, operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated. No new failure mode is introduced due to the administrative changes since the proposed changes do not involve the addition or modification of equipment nor do they alter the design or operation of affected plant systems, structures, or components. Therefore, operation of either facility in accordance with its proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The operating limits and functional capabilities of the affected systems, structures, and components are unchanged by the proposed amendments. The proposed changes to add the MDS position have management and administrative controls associated with the required qualification requirements. The St. Lucie Unit 1 and Unit 2 Technical Specifications will ensure that any individual filling the MDS position has the requisite education, experience, and training. The proposed changes do not alter the basis for any technical specification that is related to the establishment of, or the maintenance of, a nuclear safety margin. Therefore, operation of either facility in accordance with its proposed amendment would not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

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

NRC Project Direct: Cecil O. Thomas.

GPU Nuclear Inc. Et Al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: September 3, 1998.

Description of amendment request: The amendment would revise Technical Specifications 3.4.A.10.e and 3.5.a.2.e to incorporate a Condensate Storage Tank level of greater than 35 feet.

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

The proposed change does not alter the design or function of any structures, systems or components and does not affect any of the parameters or conditions that could contribute to initiation of any accidents.

The proposed change eliminated an inconsistency between the noted tank level and required water volume and, thereby, ensures 360,000 gallons of water are available for use. The proposed change does not affect the volume of water required to be available, the conditions under which it must be available nor the manner in which it will be used. Therefore, the proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Eliminating an inconsistency between the noted tank level and the required water volume does not alter the designs or function of any structures, systems or components. The proposed tank level requirement is within the design parameters of the tank and, as such, does not [ ] introduce any new mechanisms which could contribute to the creation of a new or different kind of accident than previously evaluated.

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

The proposed change eliminates an inconsistency between the noted tank level and required water volume. The proposed change ensures that an adequate makeup source is available and, in addition, that sufficient water volume is available to support operation of the core spray system in the event of a reactor vessel leak. Therefore, the proposed TS 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.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753

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

NRC Project Director: William M. Dean.

Niagara Mohawk Power Corporation, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1. (NMP1) Oswego County, New York

Date of amendment request: December 30, 1998.

Description of amendment request: The footnote of current Technical Specification (TS) Table 3.6.14–2, Radioactive Gaseous Effluent Monitoring Instrumentation, specifies that the requirement for the emergency condenser system to have one operable noble gas activity monitor per vent, is applicable during reactor power operating conditions. Note (h) of current TS Table 4.6.14–2 specifies that the requirement to perform a sensor check once per day of the emergency condenser system noble gas activity monitor is applicable during reactor power operating conditions. The proposed amendment would change the footnote of TS Table 3.6.14–2 and note (h) of TS Table 4.6.14–2 to extend the applicability of the channel operability and daily sensor check surveillance requirement from during reactor power operating conditions, to during power operation conditions and whenever the reactor coolant temperature is greater than 212 °F except for hydrostatic testing with the reactor not critical. The proposed changes would also correct a clerical error in TS 4.6.15.d. The clerical error cited an incorrect TS.

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 operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes extend the application of operability and daily sensor check for the Emergency Condenser Vent Noble Gas Activity Monitors to include, in addition to power operations, the condition when reactor coolant temperature is greater than 212 °F, except for hydrostatic testing. These changes will make the conditions for Emergency Condenser Vent Noble Gas Activity Monitor operability and daily sensor check surveillance performance consistent with the conditions for ECS [emergency cooling system] operability as indicated in LCO [Limiting Condition for Operation] 3.1.3.a.

The proposed changes to the Emergency Condenser Vent Noble Gas Activity Monitor operability and daily sensor check surveillance requirements will continue to provide assurance that the intent of the effluent monitoring requirements of 10 CFR 50 Appendix A, GDC [General Design Criterion] 64, is satisfied and the radiological effluents are maintained within the dose and dose rate limits specified in 10 CFR 50 Appendix I, 10 CFR 20, and the RETS [Radiological Effluent Technical Specifications]. The proposed changes will not effect the capability of the ECS to mitigate the consequences of an accident that results in a loss of feedwater or reactor isolation from the primary heat sink and aid the Core Spray System and Automatic Depressurization System in providing effective core cooling following non-limiting small breaks.

The proposed changes also correct a clerical error in the Uranium Fuel Cycle effluent monitoring SR [surveillance requirement]. The proposed correction simply restores the SR to the form that existed before the error was introduced. The clerical error did not affect the ODCM [Offsite Dose Calculation Manual] implementing procedures or plant operation. Thus, the cumulative dose contribution from Uranium Fuel Cycle sources will continue to be maintained within the limits of 40 CFR 190 and the RETS.

Based on the above analysis, the proposed changes do not result in any hardware changes or physical alteration of the plant, and the changes will have no impact on the design or function of any structure, system or component (SSC). As such, the SSC process variables, characteristics, and functional performance will be maintained consistent with the event initiator and the initial condition assumptions for the accident analyses. Moreover, the proposed changes will not eliminate any actions or adversely affect any SSCs required to prevent accidents or mitigate accident conditions, nor will the changes result in the degradation of any fission product barriers so as to increase the radiological consequences of an accident. It is, therefore, concluded that operation in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not result in any hardware changes or physical alteration of the plant, and the changes do not impact the design or function of any SSC. The proposed changes maintain the capability of the ECS to respond to accidents, including non-limiting small breaks, consistent with the current analyses. In addition, the proposed changes provide continued assurance that the radiological dose and dose rates will be maintained within limits. The proposed changes do not alter the process variables, characteristics, or functional performance of any SSC, do not eliminate any requirements, and do not impose any new requirements which could introduce new equipment failure modes or create new credible accidents. It is, therefore, concluded that operation in accordance with proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Nine Mile Point Unit 1, in accordance with the proposed

amendment, will not involve a significant reduction in a margin of safety.

The proposed changes do not affect the capability of the ECS to mitigate consequences of an accident that results in a loss of feedwater or reactor isolation from the primary heat sink, or affect the capability of the ECS to aid the Core Spray System and the Automatic Depressurization System in providing effective core cooling following non-limiting small breaks. Thus, there will be no impact on the post-accident radioactive material release analyses or a reduction in the margin to the associated 10 CFR 100 dose limits. In addition, the proposed changes provide continued assurance that the intent of the effluent monitoring requirements of 10 CFR 50 Appendix A, GDC 64, is satisfied and the dose and dose rates due to the radiological effluents are maintained within the limits specified in 10 CFR 50 Appendix I, 10 CFR 20, 40 CFR 190, and the RETS. Moreover, the proposed changes do not eliminate any requirements or responsibilities, nor impose new requirements or responsibilities, or alter any physical parameters which could reduce the margin to an acceptance limit. It is, therefore, concluded that operation in accordance with the proposed amendment 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

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

NRC Project Director: S. Singh Bajwa, Director.

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

Date of amendment request: December 16, 1998.

Description of amendment request: The proposed editorial and administrative changes to the Technical Specifications would either revise references and statements that are inaccurate or provide relief from administrative controls which provide insignificant safety benefit.

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 design basis accidents are not affected by the proposed editorial and administrative changes. The proposed changes do not change the level of programmatic controls or the procedural details currently in place. The proposed changes do not revise the station design, the response of the station to transients nor the manner in which the station is operated, therefore, these changes have no adverse affect to the safe operation of the station. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. There are no changes to the source term, containment isolation or radiological release assumptions used in evaluating the radiological consequences in the Seabrook Station UFSAR. Existing system and component redundancy is not being changed by the proposed changes. The proposed changes have no adverse affect on component or system interactions. The proposed changes are editorial and administrative in nature and do not change the level of programmatic controls and procedural details associated with the aforementioned technical specifications. Therefore, since there are no changes to the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated and surveilled, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

There are no changes being made to the Technical Specification safety limits or safety system settings that would adversely affect plant safety. The changes do not affect the operation of structures, systems or components nor do they introduce administrative changes to plant procedures that could affect operator response during normal, abnormal or emergency situations. 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Project Director: William M. Dean.

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

Date of amendment request: December 16, 1998.

Description of amendment request: The proposed change would relocate Technical Specifications (TS) 3/4.7.10, ''Area Temperature Monitoring,'' and associated TS Table 3.7–3, to the Seabrook Station Technical Requirements Manual.

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 change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. The proposed change does not alter or prevent the ability of structures, systems, or components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR). The proposed change is administrative in nature and does not decrease the effectiveness of programmatic controls or the procedural details of assuring operation of the facility in a safe manner.

The provisions of TS 3/4.7.10 for area temperature monitoring of the referenced selected areas is neither part of an initial condition of a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier, nor is area temperature monitoring relied upon as a primary success path to mitigate such events. The provisions for area temperature monitoring is not related to events that are considered frequent or dominant contributors to plant risk. Area temperature monitoring is not considered a design feature or an operating restriction that is an initial condition of a design basis accident or transient analysis, nor does it provide a function or actuate any accident mitigation feature in order to mitigate the consequences of a design basis accident or transient.

Relocating TS 3/4.7.10 to the Technical Requirements Manual will still provide adequate controls for area temperature in those areas designated in TS Table 3.7–3. The relocated requirements of TS 3/4.7.10 to the Technical Requirements Manual will continue to be administratively controlled in accordance with TS Section 6.0, ''Administrative Controls.''

The Seabrook Station Technical Requirements Manual is a licensee-controlled document which contains certain technical requirements and is the implementing manual for the Technical Specification Improvement Program. Changes to these requirements are reviewed and approved in accordance with Seabrook Station Technical Specifications, Section 6.7, and as outlined in the Technical Requirements Manual. Specifically, changes to the Technical Requirements require a 10 CFR 50.59 safety evaluation and are reviewed and approved by the Station Operations Review Committee (SORC) and the Nuclear Safety Audit Review Committee (NSARC) prior to implementation.

The proposed change will not degrade the ability of systems, structures and components important to safety to perform their safety function. The proposed change will not change the response of any system, structure or component important to safety as described in the Seabrook Station Updated Final Safety Analysis Report (UFSAR). Since the plant response to an accident will not change, there is no change in the potential for an increase in the consequences of an accident previously analyzed. As such, 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 previously analyzed.

The proposed change does not alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. There are no changes to the source term, containment isolation or radiological release assumptions used in evaluating the radiological consequences in the Seabrook Station UFSAR. Existing system and component redundancy is not being changed by the proposed change. The proposed change has no adverse impact on component or system interactions. The proposed change will not adversely degrade the ability of systems, structures and components important to safety to perform their safety function nor change the response of any system, structure or component important to safety as described in the Seabrook Station Updated Final Safety Analysis Report (UFSAR). The proposed change is administrative in nature and does not change the level of programmatic controls and procedural details controls of assuring operation of the facility in a safe manner. Therefore, since there are no changes to the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated and surveilled, the proposed change does not create the possibility of a new or different kind of accident from any previously analyzed.

Future changes to area temperature monitoring requirements will be reviewed and approved in accordance with Seabrook Station Technical Specifications, Section 6.7, and as outlined in the Technical Requirements Manual. Specifically, changes to the Technical Requirements require a 10 CFR 50.59 safety evaluation and are reviewed and approved by the Station Operations Review Committee (SORC) and the Nuclear Safety Audit Review Committee (NSARC) prior to implementation.

Since the plant response to an accident will not change, there is no change in the potential for an increase in the consequences of an accident previously analyzed, nor can it create the possibility of a new or different kind of accident from any previously evaluated.

Relocation of the area temperature monitoring requirements to the Technical Requirements Manual will not create the possibility of a new or different kind of accident from any previously analyzed.

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

There is no adverse impact on equipment design or operation and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. The proposed change is administrative in nature and does not change the level of programmatic controls and procedural details associated with area temperature monitoring to ensure that environmentally qualified equipment will not be exposed to temperatures beyond that which they were originally qualified.

Future changes to the area temperature monitoring requirements will be reviewed and approved in accordance with Seabrook Station Technical Specifications, Section 6.7, and as outlined in the Technical Requirements Manual. Specifically, changes to the Technical Requirements require a 10 CFR 50.59 safety evaluation and are reviewed and approved by the Station Operations Review Committee (SORC) and the Nuclear Safety Audit Review Committee (NSARC) prior to implementation.

Relocation of the requirements contained in TS 3/4.7.10 to the Technical Requirements Manual does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Project Director: William M. Dean.

Northeast Nuclear Energy Company (NNECO), Et Al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: December 28, 1998.

Description of amendment request: NNECO is proposing to change Technical Specification 2.2.1, ''Limiting Safety System Settings—Reactor Trip Setpoints,'' and the associated Bases to reflect revised loss of normal feedwater (LONF) analyses. An additional Technical Specification Bases change to the floor value for the thermal margin low pressure reactor trip is also included. This proposed change is not related to the revised LONF analyses.

NNECO is also seeking NRC approval to incorporate changes to the Millstone Unit No. 2 Final Safety Analysis Report (FSAR). The proposed changes to the FSAR, except the floor value for thermal margin low pressure reactor trip, are associated with the revised LONF analyses.

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

In accordance with 10 CFR 50.92, NNECO has reviewed the proposed changes and has concluded that they do not involve a Significant Hazards Consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

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

The analysis of a loss of normal feedwater (LONF) event, as described in the Millstone Unit No. 2 FSAR Chapters 10 and 14, has been revised. Certain key assumptions have been changed to ensure acceptable analysis results. An evaluation of the LONF analyses changes, and associated Technical Specification changes will be presented. In addition, an evaluation of an additional non LONF analyses related Technical Specification Bases and FSAR change is included.

LONF analyses changes. The LONF analyses, contained in FSAR Chapters 10 and 14, have been revised using a steam generator liquid inventory assumption, at the time of reactor trip on low steam generator water level, that is consistent with the design of the replacement steam generators. The revised Chapter 10 and 14 LONF analyses also incorporate a reduction in auxiliary feedwater delivery rates resulting from a recalculation of the Auxiliary Feedwater (AFW) System flows. The results of revised analyses indicate that the analytical limit for the low steam generator water level reactor trip must be raised to 43% narrow range level from the current 34% narrow range level. This will result in a change to the low steam generator water level reactor trip setpoint listed in Technical Specification 2.2.1.

The revised Chapter 14 LONF analysis will now take credit for automatic initiation of the motor driven auxiliary feedwater (MDAFW) pumps. The current Chapter 14 LONF analysis assumes auxiliary feedwater flow will be initiated 10 minutes after the event. The Chapter 10 LONF analysis assumption of automatic initiation of one MDAFW pump within 4 minutes, after the low steam generator level AFW actuation setpoint is reached, has not changed.

To demonstrate that one MDAFW pump delivers sufficient flow to preclude steam generator dryout, the Chapter 10 LONF analysis will not take credit for the operation of the steam generator atmospheric dump valves, instead of the main steam safety valves as in the current analysis. This new assumption yields lower predicted steam generator pressures which result in an increase in the delivered AFW flows.

LONF analyses related technical specification changes. The trip setpoint and allowable value for the low steam generator water level reactor trip will be changed to be consistent with the revised LONF analyses. The revised analyses assume an analytical limit of 43% narrow range level, instead of the current analytical limit of 34% narrow range level. The calculation of the trip setpoint, which includes instrument uncertainty, has determined that the trip setpoint should be changed from [greater than or equal to] 36.0% to [greater than or equal to] 48.5%.

The increase in the low steam generator level Reactor Protection System (RPS) actuation setpoint from [greater than or equal to] 36% to [greater than or equal to] 48.5% will result in an increase in the probability of an RPS actuation on low steam generator water level since the difference between the proposed setpoint and the normal operating value of steam generator level will decrease. The proposed actuation setpoint is below the normal operating level of 60 to 75%. Steam generator level is not expected to approach the actuation setpoint during normal operation. An unexpected plant event (e.g., loss of main feedwater or difficulty controlling steam generator level at low power levels) would be necessary for steam generator level to approach the actuation setpoint. To provide the operators with advance notice of the steam generator low level condition, the existing RPS low steam generator water level pretrip alarm setpoint will be changed to provide approximately the same margin between pretrip and trip as currently exists (5%). This will ensure that the pretrip alarm is received prior to reaching the actual record trip setpoint. Therefore, even though the proposed change will decrease the margin between the normal operating steam generator level and the RPS actuation setpoint, this change will not significantly impact the probability of an RPS actuation on low steam generator level during normal plant operations. In addition, the proposed setpoint and allowable value change will ensure a reactor trip signal is generated at, or before the analytical limit used in the revised LONF analysis is reached. Therefore, the RPS will continue to function as designed to mitigate the consequences of the design basis accidents.

The basis for the steam generator level low reactor trip will be modified to be consistent with the revised LONF analyses. The discussion concerning available water inventory and time until auxiliary feedwater is required will be removed. The proposed change to the FSAR will include a discussion of the relationship between the LONF analysis and the need to automatically initiate auxiliary feedwater flow.

Non LONF analyses related technical specification bases and FSAR change. This Technical Specification Bases and FSAR change is not related to the revised LONF analyses.

The basis for the thermal margin low pressure (TMLP) reactor trip (Technical Specification 2.2.1 Bases) will be modified. The current basis states that the floor, or minimum value, for this trip function is set at 1850 psia pounds per square inch absolute]. This value will be changed to be consistent with instrument uncertainty calculations that have determined that the floor should be increased to 1865 psia. The increase in floor value is the result of greater instrument uncertainties when harsh containment environment conditions are included.

The increase in the TMLP floor (from 1850 psia to 1865 psia) could result in an increase in the probability of an RPS actuation on thermal margin low pressure since the difference between the proposed floor setpoint and the normal operating value of pressurizer pressure will decrease. However, the proposed actuation setpoint is significantly below the normal operating pressure of approximately 2250 psia. Pressurizer pressure is not expected to approach the actuation setpoint during normal operation. A significant plant event (e.g., loss of primary coolant) would be necessary for a rapid pressure excursion to approach the actuation setpoint. Since the setpoint change is small, it will not adversely impact the probability of an RPS actuation on low pressurizer pressure during normal plant operations. In addition the proposed change to the floor value will ensure a reactor trip signal is generated at, or before the analytical limit used in the respective accident analyses is reached. Therefore, the RPS will continue to function as designed to mitigate the consequences of the design basis accidents.

Conclusion. The results of the revised LONF analyses contained in FSAR Chapters 10 and 14 have concluded that the LONF event does not result in the violation of the Specified Acceptable Fuel Design Limits, that the peak pressurizer pressure does not exceed 110% of the design pressure, that liquid primary coolant is not expelled through the pressurizer safety valves, and that adequate cooling water is supplied by the AFW System to prevent steam generator dryout and allow a safe and orderly plant shutdown. By preventing steam generator dryout, sufficient removal of decay heat from the Reactor Coolant System (RCS) will occur, preventing excessive RCS heatup and pressurization. This will ensure the steam generator fatigue analysis remains valid, and excessive discharge of primary coolant through the pressurizer safety valves does not occur. Therefore, there will be no adverse effect on the consequences of a LONF event. This is consistent with the acceptance criteria contained in Standard Review Plan (SRP) 15.2.7, [''Loss of Normal Feedwater Flow,'' Rev. 1—July 1981]. (Millstone Unit No. 2 is not an SRP plant.)

The proposed changes do not alter the way any structure, system, or component functions. The changes in actuation setpoints and equipment used in the LONF analyses affect equipment important to the mitigation of design basis accidents. These changes do not affect any equipment that can cause a design basis accident to occur. Therefore, the proposed changes do not affect the probability of occurrence of a previously evaluated accident.

These proposed changes do not alter the way any structure, system, or component functions. There will be no adverse effect on any design basis accident previously evaluated, on any equipment important to safety, or on the radiological consequences of any design basis accident. Therefore, these proposed changes will not adversely affect the consequences of a previously evaluated accident.

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

Results of the proposed LONF analyses have demonstrated that the Specified Acceptable Fuel Design Limits are not violated, that the peak pressurizer and steam generator pressures do not exceed 110% of the design pressure, that liquid primary coolant is not expelled through the pressurizer safety valves, and that adequate cooling water is supplied by the AFW System to prevent steam generator dryout and allow a safe and orderly plant shutdown. Therefore, there are no new or different types of failures of systems or equipment important to safety which could cause a new or different type of accident from any accident previously evaluated.

The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The revised FSAR Chapter 14 analysis has concluded that the steam generator low water level reactor trip setpoint does not provide sufficient water inventory in the steam generators at the time of the reactor trip such that auxiliary feedwater flow will not be required for 10 minutes. This contradicts the current Technical Specification Basis (Technical Specification 2.2.1) for the steam generator low water level reactor trip setpoint. Therefore, the revised analysis reduces the margin of safety as defined in the Bases of the Millstone Unit No. 2 Technical Specifications. However, with the proposed changes to increase the low steam generator water level reactor trip setpoint and taking credit for automatic AFW System actuation, it has been shown that operation of these systems can mitigate the LONF event, and ensure plant response is within the acceptance criteria. Results of the proposed LONF analyses have demonstrated that the Specified Acceptable Fuel Design Limits are not violated, that the peak pressurizer and steam generator pressures do not exceed 110% of the design pressure, that liquid primary coolant is not expelled through the pressurizer safety valves, and that adequate cooling water is supplied by the AFW System to prevent steam generator dryout and allow a safe and orderly plant shutdown. Therefore, these proposed changes do not involve a significant reduction in a margin of safety.

The proposed change to the floor value for the TMLP reactor trip function is the result of a revision to the instrument loop uncertainty and setpoint calculations. The proposed change to the Technical Specification Basis will incorporate the RPS TMLP floor setpoint change. This change to the TMLP floor will not adversely affect this function. The TMLP reactor trip function will still operate as designed. The RPS will continue to function as designed to mitigate the consequences of design basis accidents. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

The NRC has provided guidance concerning the application of standards in 10 CFR 50.92 by providing certain examples (March 6, 1986, 51 FR 7751) of amendments that are considered not likely to involve an SHC. The changes proposed herein are not enveloped by any specific example.

As described above, this License Amendment Request does not impact the probability of an accident previously evaluated, does not involve a significant increase in the consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident from any accident previously evaluated, and does not result in a significant reduction in a margin of safety. Therefore, NNECO has concluded that the proposed changes do not involve an SHC.

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

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

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

NRC Project Director: William M. Dean.

Northeast Nuclear Energy Company (NNECO), Et Al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: January 18, 1999.

Description of amendment request: NNECO is proposing to change Technical Specification 3.6.1.2, ''Containment Systems—Containment Leakage.'' The Bases for this Technical Specification and the Final Safety Analysis Report (FSAR) will also be modified to address the proposed changes.

The limit for secondary containment bypass leakage specified in Technical Specification 3.6.1.2.c will be reduced from less than 0.017 L_a to less than 0.0072 L_a. This new limit is consistent with the value of secondary containment bypass leakage used in the revised off-site and control room dose calculations following a design basis loss-of-coolant accident (LOCA).

Technical Specification 3.6.1.2.c will be modified by replacing ''identified in Table 3.6–1 as'' with ''that are.'' This will allow Table 3.6–1 to be removed. The removal of this table from Technical Specifications and the proposed wording change are consistent with the guidance contained in Generic Letter (GL) 91–08. It is not necessary to maintain a list of the secondary containment bypass leakage paths in Technical Specifications. The Millstone Unit No. 2 FSAR (Section 5.3.4) provides the necessary information to determine the secondary containment bypass leakage paths that must be considered to ensure that the combined leakage rate limit contained in Technical Specification 3.6.1.2.c is met.

Technical Specification 3.6.1.2 Table 3.6–1, ''Secondary Containment Bypass Leakage Paths,'' will be removed and the phrase ''This Page Intentionally Deleted'' will be added to Page 3/4 6–5.

The Bases for Technical Specification 3.6.1.2 will be modified to indicate that the Millstone Unit No. 2 FSAR contains a list of the containment penetrations that have been identified as secondary containment bypass leakage paths.

FSAR Section 5.3.4, ''Through-Line Leakage Evaluation,'' will be changed to include the additional secondary containment bypass leakage paths that have been identified. The criteria used to determine the secondary containment bypass leakage paths will be modified to be consistent with the criteria used in the evaluation that identified the additional leakage paths.

The discussion of the use of a leakage rate of 11 cc/hr for the control room dose calculations will be modified. The revised control room dose calculations will assume a total secondary containment bypass leakage rate consistent with the proposed change to Technical Specification 3.6.1.2.

As a result of these proposed changes, the calculated off-site and control room doses following a design basis LOCA will change. The calculated doses are specified in FSAR Section 14.8.4, ''Radiological Consequences of the Design Basis Accident.'' A revision to this section of the FSAR has been submitted to the NRC by the letter dated September 28, 1998. This submittal will be revised to incorporate the proposed total secondary containment bypass leakage rate and the associated change to the calculated off-site and control room doses following a design basis LOCA.

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

In accordance with 10 CFR 50.92, NNECO has reviewed the proposed changes and has concluded that they do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

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

The proposed change to lower the limit for secondary containment bypass leakage, as specified in Technical Specification 3.6.1.2.c, from [less than] 0.017 L_a to [less than] 0.072 L_a will reduce the off-site doses associated with the design basis LOCA. The proposed change to raise the limit for secondary containment bypass leakage from 11 cc/hr to [less than] 0.0072 L_a will increase the dose to the Control Room Operators following a design basis LOCA. However, the revised off-site and control room dose calculations, using the proposed combined secondary containment bypass leakage limit, demonstrate that the limits of 10 CFR 100 and 10 CFR 50, Appendix A, General Design Criteria (GDC) 19 are met. In addition, these proposed changes will result in the use of the same limit for secondary containment bypass leakage when determining the radiological consequences of a design basis LOCA.

The proposed wording change to Technical Specification 3.6.1.2.c, and the associated removal of Table 3.6–1, will not change the requirement to verify total secondary containment bypass leakage is within the limit assumed in the determination of the radiological consequences of the design basis LOCA. Control of the penetrations that have been identified as secondary containment bypass leakage paths will be maintained by the process used to change the Millstone Unit No. 2 FSAR. This process ensures that appropriate changes to the FSAR are evaluated in accordance with 10 CFR 50.59 to determine if NRC approval is required prior to implementing the change. This process also ensures that the NRC is informed of FSAR changes via regular updates to the FSAR. The removal of Table 3.6–1 from Technical Specifications and the proposed wording change are consistent with the guidance contained in GL 91–08.

The identification and addition of more secondary containment bypass leakage paths to the FSAR will have no impact on the calculated off-site and control room doses following a design basis LOCA since the combined leakage through all secondary containment bypass leakage paths is limited to the proposed value contained in Technical Specification 3.6.1.2. The addition of bypass leakage paths does not change the combined leakage limit, which is now used in the off-site and control room dose calculations.

The Bases for Technical Specification 3.6.1.2 will be modified to indicate that the Millstone Unit No. 2 FSAR contains a list of the containment penetrations that have been identified as secondary containment bypass leakage paths.

The proposed changes do not alter the way any structure, system, or component functions. These changes do not affect any equipment that can cause a design basis accident to occur. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. The reduction in the allowable secondary containment bypass leakage limit will result in a decrease in the calculated off-site doses associated with the design basis LOCA. The use of the proposed secondary containment bypass leakage limit will increase the calculated doses to the Control Room Operators following a design basis LOCA. However, the calculated doses meet the criteria of 10 CFR 100 and GDC 19. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. Also, the response of the plant and the operators following these accidents is essentially unaffected by the change. The criteria used by the plant operators to terminate containment spray following a design basis LOCA will change from containment pressure to either time or pressure, whichever requires longer operation. This will ensure that containment spray remains in operation long enough to achieve the assumed iodine decontamination. However, the operator action to terminate containment spray will remain the same. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change to lower the Technical Specification limit for secondary containment bypass leakage, to remove Table 3.6–1, and to add more secondary containment bypass leakage paths to the FSAR will have no adverse effect on equipment important to safety. The equipment will continue to function as assumed in the design basis accident analysis. These changes will ensure that the secondary containment bypass leakage paths are identified and tested to verify that the total secondary containment bypass leakage does not exceed the Technical Specification limit. This will ensure that the expected off-site and control room doses following a design basis LOCA are within the limits specified in 10 CFR 100 and GDC 19. Therefore, there will be no significant reduction in the margin of safety as defined in the Bases for the Technical Specification affected by these 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.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

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

NRC Project Director: William M. Dean.

Northeast Nuclear Energy Company (NNECO) Et Al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: January 18, 1999.

Description of amendment request: The proposed changes will remove the Technical Specification related to Hydrogen Purge System from the Millstone Unit No. 2 Technical Specifications. The proposed changes affect Technical Specifications 3/4.6.4.3, ''Containment Systems, Hydrogen Purge System.'' The Bases of the associated Technical Specification will be modified to address the proposed changes. The proposed changes will allow the licensee to downgrade the hydrogen purge system to a non-safety-related system.

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

In accordance with 10 CFR 50.92, NNECO has reviewed the proposed changes and has concluded that they do not involve a Significant Hazards Consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

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

The Hydrogen Purge System provides a backup means to manually control the hydrogen concentration in containment given the multiple failure of the redundant, Seismic Category I Hydrogen Recombiner System. The primary success path for hydr9gen control is the Hydrogen Recombiner System. The Hydrogen Recombiner System has redundant trains and is fully qualified to maintain hydrogen control following a design basis accident. FSAR [Final Safety Analysis Report] Section 14.8.3.5, ''Radiological Consequences of Purging'' is being removed from the FSAR since it is no longer required. Since the hydrogen recombiners are fully redundant, it is not necessary to postulate offsite doses for purge during a design basis accident. Thus, the deletion of consequences does not represent a change in the consequences of a design basis event. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

Revision of Index Page VII is an administrative change. The proposed change to Bases section 3/4.6.4 by deleting reference to ''the purge system'' is required since Technical Specification 3/4.6.4.3 is being removed. Therefore, these changes will not significantly increase the probability or consequences of an accident previously evaluated.

The proposed changes do not alter how any structure, system, or component functions. There will be no effect on equipment important to safety. The proposed changes have no effect on any of the design basis accidents previously evaluated. Therefore, this License Amendment Request does not impact the probability of an accident previously evaluated, nor does it involve a significant increase in the consequences of an accident previously evaluated.

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

The purge system is a standby purge system which is not in service during normal operations as a hydrogen purge system (i.e., Charcoal Filter Heaters de-energized). Therefore, no new accident is created either by system unavailability or actuation. The FSAR will still address the use of the purge system as a backup to the recombiner system, Revision of Index Page VII is an administrative change. The proposed change to Bases section 3/4.6.4 by deleting reference to ''the purge system'' is required since Technical Specification 3/4.6.4.3 is being removed. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety is defined in the Bases 3/4.6.4 which states that the ''hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7 * * *''. Regulatory Guide 1.7 describes methods that would be acceptable in meeting the standards for a combustible gas control system, 10 CFR 50.44, ''Standards for combustible gas control systems in light-water-cooled power reactors.'' Regulatory Guide 1.7 acknowledges that purging is a means of reducing the hydrogen concentration but it should not be the primary means because of the release of radioactivity to the environment. The regulatory guide does advise that there be an ''installed capability for a controlled purge of the containment atmosphere to aid in cleanup.'' Removal of the Hydrogen Purge System Technical Specification is consistent with Regulatory Guide 1.7. Additionally, the capability to purge is still documented in the FSAR. Revision of Index Page VII is an administrative change. The proposed change to Bases section 3/4.6.4 by deleting reference to ''the purge system'' is required since Technical Specification 3/4.6.4.3 is being removed. Therefore, the proposed changes will not result in a significant reduction in the margin of safety as defined in the Bases for Technical Specifications covered in this License Amendment Request.

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

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

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

NRC Project Director: William M. Dean.

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: January 18, 1999.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) 3/4.2.2 to be in accordance with NRC-approved Westinghouse methodologies for the heat flux hot channel factor—$F_Q(Z)$. In addition, the proposed amendment would make changes to the core operating limits and the analytical methods used to determine core operating limits contained in Section 6.9.1.6.a and b, respectively, by adding, modifying, or deleting references.

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 signification hazards consideration, which is presented below:

NNECO has reviewed the proposed revision in accordance with 10 CFR 50.92 and has concluded that the revision does not involve any Significant Hazards Considerations (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not satisfied. The proposed Technical Specification revision does not involve an SHC because the revision would not:

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

To determine any potential impact, the proposed changes to the TS are grouped into the following two categories.

(a) Changes to Technical Specification 3/4.2.2 ''Heat Flux Hot Channel Factor—$F_Q(Z)$''

(b) Changes that are not related to the Heat Flux Hot Channel Factor TS, and are administrative in nature. These include defining a new core operating limit and deleting, re-numbering, updating and adding references to analytical methods used to determine core operating limits in TS 6.9.1.6 ''Core Operating Limit Report (COLR).['']

With respect to item 1.a changes related to the Heat Flux Hot Channel Factor, $F_Q(Z)$, impact the initial conditions assumed in the accidents analyzed for MP3 [Millstone Unit 3]. These initial conditions are power distributions which are consistent with reactor operation as defined in the TS. The proposed changes to the Heat Flux Hot Channel Factor TS ensure that proper actions are taken to maintain peaking factors within the limits assumed in the MP3 accident analysis. The proposed changes are consistent with the NRC approved Westinghouse methodology for $F_Q(Z)$ surveillance. Changes to the SURVEILLANCE and ACTION statements will not change the probability of occurrence of any analyzed accidents. Furthermore, the consequences of analyzed accidents will not change since the power distribution assumptions will not be challenged by reactor operation allowed by the Technical Specifications.

With respect to item 1.b the administrative changes to the Technical Specifications do not affect existing or proposed Limiting Conditions for Operation (LCO) or SURVEILLANCE REQUIREMENTS. Therefore, there is no impact on the design basis accidents.

Thus it is concluded that the proposed revision 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.

(a) Proposed changes to the Heat Flux Hot Channel Factor, TS 3/4.2.2 ensure that proper actions are taken to maintain peaking factors within the limits assumed in the MP3 accident analysis. The proposed changes are consistent with the NRC approved Westinghouse methodology for $F_Q(Z)$ surveillance. Maintaining safety analysis assumptions on power distributions cannot be an initiating event for any design basis accidents and will not create the possibility of a different type of accident. Therefore the changes associated with the Heat Flux Hot Channel Factor limiting condition for operation do not represent a new unanalyzed accident.

(b) Since the administrative changes do not affect plant operation, the potential for an unanalyzed accident is not created. No new failure modes are introduced.

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

3. Involve a significant reduction on the margin of safety.

(a) The proposed changes ensure that $F_Q(Z)$, will remain within the safety analysis assumptions. The LCO limits and SURVEILLANCE REQUIREMENTS are not altered. Therefore, the impact on the consequences on the protective boundaries is unchanged. Meeting the intent of the NRC approved Westinghouse methodology for $F_Q(Z)$, SURVEILLANCE ensures that power distributions assumed in the accident analysis will not be challenged by reactor operations allowed by the Technical Specifications. Therefore, verification of no change in the margin of safety is encompassed by meeting the power distribution limits assumed in analyzed accidents.

(b) Since the proposed changes do not affect the consequences of any accident previously analyzed, there is no reduction in the margin of safety.

Thus it is concluded that the proposed revision does not involve a significant reduction in the margin of safety.

In conclusion, based on the information provided, it is determined that the proposed revision does not involve a Significant Hazard 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.

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

NRC Project Director: William M. Dean.

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

Date of amendment request: December 31, 1998.

Description of amendment request: The proposed amendment would revise the technical specification (TS) reactor pressure vessel (RPV) pressure-temperature (P–T) limit curves, delete completed RPV sample surveillance requirements, delete requirement to withdraw a specimen at next refueling outage, and remove the standby liquid control system (SBLC) relief valve setpoint. Associated administrative changes are also proposed.

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

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

RPV P–T curve changes. It is proposed that P–T curves be revised to accommodate the shift in RT_{NDT} determined using actual surveillance program data rather than generic data provided in Regulatory Guide [RG] 1.99 Revision 2 (Radiation Embrittlement of Reactor Vessel Materials). The new P–T curves will increase the margins provided in the P–T limit curves against non-ductile failure of the RPV. Regulatory Guide 1.99 Revision 2 encourages use of plant specific surveillance data as data becomes available.

Eliminating prescriptive requirements to remove a RPV test specimen sample at three fourths service life will result in an overall improvement in the RPV surveillance program since the limited number of remaining surveillance samples will be removed at optimum intervals. Therefore, proposed changes will neither significantly increase the probability or the consequences of an accident previously evaluated.

RPV surveillance requirements. Deleting completed, one time surveillance requirements [SRs] of SR section 4.6.B and incorporating a discussion of the results in the Bases is an administrative change and has no effect on probability or consequences of accidents.

SBLC relief valve setpoint testing. The testing requirements of TS section 4.4.A.2.c are enveloped by the current testing performed by Monticello's IST [inservice test] Program, which implements ASME [American Society of Mechanical Engineers] Code Section XI, approved by 10 CFR 50.55a. The IST program requires all relief valves to be tested to their nameplate data setpoints. Any modification to a relief valve's nameplate data is controlled by the plant's configuration control process which would ensure the requirements of ASME Section XI are invoked as required by TS section 3.15. The IST program required by TS 4.15 ensures the SBLC relief valves will be properly tested for operability. Therefore, revising section 4.4.A.2.c to remove specific setpoints does not increase the probability or consequences of an accident.

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

RPV P–T curve change. Updated RPV P–T limit curves will not create the possibility of a new or different kind of accident nor alter operational standards. New limits continue a system of operating bounds which are in place to prevent damage to reactor vessels during normal operating conditions including hydrostatic pressure and leakage testing, and anticipated transients. The updated P–T curves incorporate the results of RPV surveillance specimen testing utilizing criteria defined in RG 1.99, Revision 2. No change is being made to the way the P–T limits provide plant protection. No new modes of operation are involved. The changes do not necessitate physical alteration of the plant.

RPV surveillance requirements. Deleting completed, one time surveillance requirements of section 4.6B and incorporating a discussion of the results in the Bases is an administrative change and therefore has no effect on previously analyzed accidents.

SBLC Relief Valve Setpoint Testing. The testing requirements of TS section 4.4.A.2.c are enveloped by the current testing performed by Monticello's IST Program, which implements ASME Code Section XI, approved by 10 CFR 50.55a. The IST program requires all relief valves to be tested to their nameplate data setpoints. Any modification to a relief valve's nameplate data is controlled by the plant's configuration control process which would ensure the requirements of ASME Section XI are invoked as required by TS section 3.15. The IST program required by TS 4.15 ensures the SBLC relief valves will be properly tested for operability. Therefore, revising section 4.4.A.2.c to remove specific setpoints does not create the possibility of a new or different kind of accident, from any accident previously analyzed.

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

RPV P–T curve change. The proposed RPV P–T curve changes are designed to maintain the recommended safety factors specified in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G, and 10 CFR Part 50, Appendix G. The revised curves are based on current NRC guidelines utilizing actual RPV surveillance program tests results. The proposed changes shift the curves in a slightly more conservative direction thus maintaining or increasing the previous margins of safety.

RPV surveillance requirements. Deleting completed, one time surveillance requirements from Section 4.6.B and incorporating a discussion of the results in the Bases is an administrative change and has no effect on any margin of safety.

SBLC relief valve setpoint testing. The testing requirements of TS section 4.4.A.2.c are enveloped by the current testing performed by Monticello's IST Program, which implements ASME Code Section XI, approved by 10 CFR 50.55a. The IST program requires all relief valves to be tested to their nameplate data setpoints. Any modification to a relief valve's nameplate data is controlled by the plant's configuration control process which would ensure the requirements of ASME Section XI are invoked as required by TS section 3.15. The IST program required by TS 4.15 ensures the SBLC relief valves will be properly tested for operability. Therefore, revising section 4.4.A.2.c to remove specific setpoints will not reduce the margin of safety.

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

Local Public Document Room location: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

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

NRC Project Director: Cynthia A. Carpenter.

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

Date of amendment request: January 4, 1999.

Description of amendment request: PECO Energy Company (PECO Energy) is requesting Technical Specifications (TS) changes which will revise the Administrative Section of TS pertaining to controlled access to High Radiation Areas, and the reporting dates for the Annual Occupational Radiation Exposure Report and the Annual Radioactive Effluent Release Report.

The specific TS changes are as follows:

TS Section 6.12, 6.12.1, and 6.12.2 will be changed to: clarify requirements; incorporate additional monitoring options (to allow dosimetry and video monitoring) for entry into high radiation areas; add the requirement that all individuals entering a high radiation area have knowledge of the dose rates in the area; and add the requirement that locked high radiation controls apply to each individual entering the area.

TS Sections 6.9.1.4, 6.9.1.5(a), and 6.9.1.8 will be changed to: support changes to the NRC reporting dates;

reference 10 CFR 20.2206; delete current reporting dates, and correct a typographical error.

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

The changes are administrative in nature and do not impact the operation, physical configuration, or function of plant equipment or systems. The changes do not impact the initiators or assumptions, of analyzed events, nor do they impact mitigation of accidents on transient events. Therefore, these changes do not increase the probability of occurrence of consequences of an accident previously evaluated in the SAR [Safety Analysis Report].

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

The proposed changes are administrative in nature and do not alter plant configuration, require that new equipment be installed, alter assumptions made about accidents previously evaluated, or impact the operation or function of plant equipment. Therefore, these changes do not create the possibility of a new or different kind of accident than previously evaluated.

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

The proposed changes are administrative in nature and do not impact any safety assumptions, or potentially reduce any margin of safety as described in the LGS TS basis. The proposed changes have no impact on any safety analysis assumptions. Therefore, these changes do not involve any reduction in a margin of safety.

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

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

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

NRC Project Director: William M. Dean.

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

Date of amendment request: December 28, 1998.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to permit an increase in the allowable leak rate for the main steam isolation valves (MSIVs) and to delete the MSIV Sealing System. The main steam drain lines and the main condenser would be utilized as an alternate MSIV leakage treatment method.

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

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

The proposed changes to TS Section 3.6.1.2 do not involve a change to structures, components, or systems that would affect the probability of an accident previously evaluated in the Hope Creek Updated Final Safety Analysis Report (UFSAR).

The proposed changes involve eliminating the Main Steam Isolation Valve (MSIV) Steam Sealing System requirements from the TS. As described in Section 6.7 of the UFSAR, the MSIV Steam Sealing System is manually initiated in about 20 minutes following a design basis Loss of Coolant Accident (LOCA). Since the MSIV Steam Sealing System is operated only after an accident has occurred, these proposed changes have no effect on the probability of an accident. Since MSIV leakage and operation of the MSIV Steam Sealing System are included in the radiological analysis for the design basis LOCA as described in Section 15.6.5 of the UFSAR, the proposed changes will not affect the precursors of other analyzed accidents. Analysis of the affects of the proposed changes do, however, result in acceptable radiological consequences for the design basis LOCA previously evaluated in Section 15.6.5 of the UFSAR.

Hope Creek has an inherent MSIV leakage treatment capability as discussed below. [Public Service Electric and Gas Company] PSE&G proposes to use the drain lines associated with the main steam lines and main turbine condenser as an alternative to the guidance in Regulatory Guide 1.96, ''Design of Main Steam Isolation Valve Leakage Control System For Boiling Water Nuclear Power Plants,'' Revision 0, May 1975, for MSIV leakage treatment. If approved, PSE&G will incorporate this alternative method in the appropriate operational procedures and Emergency Operating Procedures.

The Boiling Water Reactor Owner's Group (BWROG) has evaluated the availability of main steam system piping and main condenser alternate pathways for processing MSIV leakage, and has determined that the probability of a near coincident LOCA and a seismic event is much smaller than for other plant safety risks. Accordingly, this proposed MSIV leakage treatment pathway will be available during and after a LOCA. Nevertheless, the BWROG has also determined that the design requirements applied to the Hope Creek main steam system piping and main condenser contain substantial margin, based on the original design requirements.

In order to further justify the capability of the main steam piping and main condenser alternate treatment pathway, the BWROG has reviewed limited earthquake experience data on the performance of non-seismically designed piping and condensers during past earthquakes. As summarized in General Electric (GE) Report, ''BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems,'' NEDC–31858P, Revision 2, submitted to the [U.S. Nuclear Regulatory Commission] NRC by BWROG letter dated October 4, 1993, this study concluded that the possibility of a failure that could cause a loss of steam or condensate in Boiling Water Reactor (BWR) main steam piping or condensers in the event of a design basis (i.e., safe shutdown) earthquake is highly unlikely, and that such a failure would also be contrary to a large body of historical earthquake experience data, and thus unprecedented.

PSE&G has performed a verification of seismic adequacy of the Hope Creek main stream piping and main condenser consistent with the guidelines discussed in NEDC–31858P, Revision 2, to provide reasonable assurance of the structural integrity of these components. This evaluation, ''Hope Creek Nuclear Plant Main Steam Isolation System Alternate Leakage Treatment Pathway Seismic Evaluation,'' clearly demonstrates that the MSIV leakage treatment drain pathway meets the intent of 10 CFR 100 Appendix A, with regards to seismic qualification. Except for the requirement to establish a proper flow path from the MSIVs to the condenser, the proposed method is passive and does not require any additional logic control and interlocks. The method proposed for MSIV leakage treatment is consistent with the philosophy of protection by multiple barriers used in containment design for limiting fission product release to the environment.

A plant-specific radiological analysis has also been performed in accordance with NEDC–31858P, Revision 2, to assess the effects of the proposed increase to the allowable MSIV leakage rate in terms of Main Control Room (MCR) and off-site doses following a postulated design basis LOCA. This analysis utilizes the hold-up volumes of the main steam piping and condenser as an alternate method for treating the MSIV leakage. As discussed earlier, there is reasonable assurance that the main steam piping and condenser will remain intact following a design basis earthquake. The radiological analysis uses standard conservative assumptions for the radiological source term consistent with Regulatory Guide (RG) 1.3, ''Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-Of-Coolant Accident for Boiling Water Reactor,'' Revision 2, dated April 1974.

The analysis results demonstrate that dose contributions from the proposed MSIV leakage rate limit of 200 scfh per steam line, not to exceed a total of 400 scfh for all four main steam lines, and from the proposed deletion of the MSIV Steam Sealing System, result in an acceptable increase to the LOCA doses previously evaluated against the regulatory limits for the off-site doses and MCR doses contained in 10 CFR 100 and 10 CFR 50, Appendix A, General Design Criterion (GDC) 19, respectively. However, the calculation methodology for the revised dose exposures were performed in a manner that included more conservative design basis assumptions (e.g., inclusion of system response times, and increased allowable leakage rates) than in the existing Hope Creek licensing basis.

The whole body doses at the low population zone (LPZ) outer boundary and MCR increase from about 0.2 rem to 0.6 rem and from 0.04 rem to 0.09 rem, respectively. These increases are not significant since the revised doses are small fractions of the regulatory limits of 25 rem and 5 rem, respectively. The associated whole body dose at the exclusion area outer boundary (EAB) increases from about 1.3 rem to 2.6 rem, which is well within the regulatory limit of 25 rem. The revised thyroid dose at the LPZ outer boundary increases from about 18 rem to 36 rem, which is well within the regulatory limit of 300 rem. The revised thyroid dose at the EAB decreases from about 175 rem to 121 rem (due to plate out on the steam piping and condenser), which is within the regulatory limit of 300 rem. However, the MCR thyroid dose increases from about 0.3 rem to 5.0 rem, which is well within the regulatory limit of 30 rem. Additionally, the MCR beta skin dose increases from about 0.9 rem to 1.6, which is well within the regulatory limit of 30 rem.

The resulting revised thyroid doses discussed above are dominated by the inorganic radioactive iodine fractions of the accident source term used in this analysis. More than 95% of the initial radioactive iodine inventory is assumed to be in the form of inorganic species in accordance with the guidance in Regulatory Guide 1.3. However, NUREG–1465, ''Accident Source Terms for Light-Water Nuclear Power Plants,'' identifies that at least 95% of the iodine entering containment would be in the form of particulate iodine. Accordingly, the calculated doses discussed above are considered to be highly conservative relative to realistic radiological source terms resulting from a postulated LOCA.

In summary, the proposed changes discussed above do not result in a significant increase in the radiological consequences of a LOCA when the same assumptions and methods specified in the UFSAR are used, recognizing that radiological consequences calculated in the UFSAR and for these proposed changes are significantly higher than those using more realistic assumptions and methods. Nevertheless, the calculated off-site and MCR doses resulting from a LOCA remain well below the regulatory limits. Although the revised LOCA doses are higher for low MSIV leakage rates, the effectiveness of the proposed alternate treatment method, even for leakage rates greater than the proposed increase in the MSIV allowable leak rate, ensures that off-site and MCR dose limits are not exceeded.

The proposed change to TS Table 3.6.3–1 involves the deletion of MSIV Steam Sealing valves and associated main steam line drain valves from the list of primary containment isolation valves. This proposed change is consistent with the proposed deletion of the MSIV Steam Sealing System. The MSIV Steam Sealing System lines and main steam line drain valves that are connected to the main steam piping will be welded and/or capped closed to assure primary containment integrity is maintained. The welding and post weld examination procedures will be in accordance with American Society of Mechanical Engineers (ASME) Code, Section III requirements. These welds and/or caps will be periodically tested as part of the Containment Integrated Leak Rate Test (CILRT). This proposed change does not involve an increase in the probability of equipment malfunction previously evaluated in the UFSAR. This proposed change has no effect on the consequences of an accident since the MSIV Steam Sealing lines and associated main steam line drain valves will be welded an/or cap closed, thus assuring that the containment integrity, isolation, and leak test capability are not compromised.

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

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

Although the proposed changes will introduce and take credit for a new level of operational performance for existing plant systems and components that have not been previously evaluated in the accident analysis, the affect on this equipment has been evaluated and found to provide an acceptable level of reliability that will provide the required level of protection. This conclusion is based on the evaluation performed in NEDC–31858P, Revision 2, and the seismic evaluation of the proposed MSIV leakage treatment pathway. Therefore, reliance on different equipment than previously assumed to mitigate the consequences of an accident does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The BWROG evaluated MSIV performance and concluded that MSIV leakage rates up to 200 scfh per line will not inhibit the capability and isolation performance of the MSIVs to effectively isolate the primary containment. Implementation of the proposed changes will not result in modifications that could adversely impact the operability of the MSIVs. The LOCA has been analyzed using the main steam piping and main condenser as a treatment method to process MSIV leakage at the proposed maximum rate of 200 scfh per main steam line, not to exceed 400 scfh total for all four main steam lines. Therefore, the proposed change to increase the allowed MSIV leakage rate does not create any new or different kind of accident from any accident previously evaluated.

The proposed change to eliminate the MSIV Steam Sealing System does not create the possibility of a new or different kind of accident from any accident previously evaluated because the removal of the MSIV Steam Sealing System does not affect any of the remaining Hope Creek systems, and the LOCA has been re-analyzed using the proposed alternate method to process MSIV leakage. The associated proposed change to delete the MSIV Steam Sealing isolation valves and associated main steam line drain valves from TS Table 3.6.3–1 does not create the possibility of a new or different kind of accident, since the affected main steam piping will be welded and/or capped closed to assure that the primary containment integrity, isolation, and leak testing capability are not compromised.

Therefore, as discussed above, the proposed changes do not create the possibility for any 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 change to TS Section 3.6.1.2 to increase the MSIV allowable leakage does not involve a significant reduction in the margin of safety. As discussed in the current Bases for TS Section 3/4.6.1.2, the allowable leak rate limit specified for the MSIVs is used to quantify a maximum amount of leakage assumed to bypass primary containment in the LOCA radiological analysis. Accordingly, results of the re-analysis supporting these proposed changes are evaluated against the dose limits contained in 10 CFR 100 for the off-site doses, and 10 CFR 50, Appendix A, GDC 19, for the MCR doses. As discussed above, sufficient margin relative to the regulatory limits is maintained even when assumptions and methods (e.g., RG 1.3) that are considered highly conservative relative to more realistic assumptions and methods, are used in the analysis.

Results of the radiological analysis demonstrate that the proposed changes do not involve a significant reduction in the margin of safety. The whole body doses, in terms of margin of safety, are insignificantly reduced by 1.6% at the LPZ, 1.0% in the MCR, and by 5.2% at the EAB. The margin of safety for thyroid doses is reduced by 6.13% at the LPZ and 15.7% in the MCR, but is actually increased by 17.3% at the EAB. The margin of safety for beta dose is insignificantly reduced by 2.4% in the MCR. These reductions in the margin of safety are not significant since the revised calculated doses are highly conservative yet remain well below the regulatory limits, and therefore a substantial margin to the regulatory limits is maintained.

Furthermore, while the proposed changes will result in a calculated reduction in the margin of safety, this reduction is not significant when considering the increased reliability and capability of the proposed MSIV leakage treatment system. The resulting revised thyroid doses discussed above are dominated by the inorganic radioactive iodine fractions of the accident source term used in this analysis. More than 95% of the initial radioactive iodine inventory is assumed to be in the form of inorganic species in accordance with the guidance in Regulatory Guide 1.3. However, NUREG–1465, ''Accident Source Terms for Light-Water Nuclear Power Plants,'' identifies that at least 95% of the iodine entering containment would be in the form of particulate iodine. Accordingly, the calculated doses discussed above are considered to be highly conservative relative to realistic radiological source terms resulting from a postulated LOCA.

The proposed change to eliminate the MSIV Steam Sealing System from TS does not reduce the margin of safety. In fact, the overall margin of safety is increased. The function of this system for MSIV leakage treatment will be replaced by alternate main steam drain lines and condenser equipment. This treatment method is effective in reducing the dose consequences of MSIV leakage over an expanded operating range compared to the capability of the MSIV Steam Sealing System and will, thereby, resolve the safety concern that the MSIV Steam Sealing System will not function at MSIV leakage rates higher than the Steam Sealing System's design capacity. Except for the requirement to establish a proper flow path from the MSIVs to the condenser, the proposed method is passive and does not require any new logic control and interlocks. This proposed method is consistent with the philosophy of protection by multiple barriers used in containment design for limiting fission product release to the environment. Furthermore, as previously identified, based on the evaluations discussed in NEDC–31858P, Revision 2, and the seismic evaluation performed for Hope Creek, the design of the MSIV leakage treatment pathway meets the intent of the 10 CFR 100, Appendix A, requirement for seismic qualification. Therefore, the proposed method is highly reliable and effective for MSIV leakage treatment.

The revised calculated LOCA doses remain within the regulatory limits for the off-site and the MCR doses. Furthermore, the revised calculation shows that MSIV leakage rates greater than 200 scfh for all four main steam lines would not exceed the regulatory limits. Therefore, the proposed method maintains a margin of safety for mitigating the radiological consequences of MSIV leakage beyond the proposed TS leakage rate limit of 200 scfh per main steam line, not to exceed a total of 400 scfh for all four main steam lines.

The proposed change to delete MSIV Steam Sealing valves from TS Table 3.6–3–1 [3.6–3–1] does not reduce the margin of safety. Welded and/or capped closure of the MSIV Steam Sealing lines assures that the primary containment integrity and leak testing capability are not compromised. These welds and/or caps will be periodically leak tested as part of the CILRT. Therefore, the proposed deletion of the MSIV Steam Sealing System isolation valves does not involve a reduction in a margin of safety.

Accordingly, based on the above reasons, 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.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070.

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

NRC Project Director: William M. Dean.

Public Service Electric & Gas Company, Docket Nos. 50–272, Salem Nuclear Generating Station, Unit No. 1, Salem County, New Jersey

Date of amendment request: January 15, 1999.

Description of amendment request: The proposed amendment would allow a one-time extension of the Technical Specification (TS) surveillance interval to the end of fuel cycle 13 for certain TS surveillance requirements (SRs). Specifically, (1) SR 4.3.2.1.3 requires the instrumentation response time and sequence testing of each engineered safety features actuation system (ESFAS) function at least once per 18 months, (2) SRs 4.8.2.3.2.f and 4.8.2.5.2.d require that the 125 volt DC and the 28 volt DC distribution system batteries, respectively, be capacity service tested at least once per 18 months, during shutdown, (3) SR 4.8.3.1.a.1.a and 4.8.3.1.a.1.b require a channel calibration and integrated system functional test for one 4.16 kilovolt reactor coolant pump circuit at least once per 18 months such that all circuits are tested at least once per 72 months, (4) SR 4.1.2.2.c requires testing to verify that each automatic valve in the reactivity control system flow path actuate on a safety injection (SI) test signal at least once per 18 months during shutdown, (50 SRs 4.3.1.1, Table 4.3–1, 4.3.2.1.1, Table 4.3–2, 4.3.3.5, Table 4.3–6, and 4.3.3.7, Table 4.3–11 require, in part, the channel calibration of pressurizer water level, pressurizer water level-high, and containment water level-wide range, the manual solid-state protection system (SSPS) functional input check, and the ESFAS manual initiation channel functional test every 18 months, (6) SR 4.5.1.d requires testing to verify each accumulator isolation valve opens automatically on an SI test signal at least once per 18 months, (7) SR 4.5.2.e.1 requires testing to verify that each automatic valve in the emergency core cooling system (ECCS) flow path actuates on an SI test signal at least once per 18 months, (8) SR 4.7.6.1.d.2 requires the control room emergency air conditioning system to automatically actuate in the pressurization mode on an SI test signal or control room intake high radiation test signal at least once per 18 months, (9) SR 4.7.10.b requires each automatic valve in the chilled water loop to actuate on an SI signal at least once per 18 months. Further, SR 4.8.1.1.2.d.7 requires a test to verify that each emergency diesel generator operates for at least 24 hours every 18 months, and SR 4.8.2.5.2.c.2 requires that the 125 volt DC battery connections be verified clean, tight, and coated with anti-corrosion material at least once per 18 months. Because of the length of the last outage and delays in restart, the SRs will be overdue prior to reaching the next refueling outage (1R13). The SRs are to be completed during the 1R13 outage, prior to returning the unit to Mode 4 (hot shutdown) upon outage completion. The proposed amendment also make some administrative and editorial changes on some of the pages that will be affected by above SR interval extensions.

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:

4.3.2.1.3 (Instrumentation, Engineered Safety Feature Actuation System Instrumentation)

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

Deferral of the surveillance requirement does not involve any physical changes to the plant nor does it change the way the plant is operated. Thus the proposal does not increase the probability of an accident previously evaluated.

The SEC [safeguard equipment control] automatic self-test feature, the monthly functional surveillance testing and the positive surveillance testing history provide sufficient assurance of the operability of the system. These features also provide assurance that a degraded condition, if it did occur, would be detected.

Thus, it is reasonable to conclude that this proposal represents no significant increase in the consequences of an accident previously analyzed.

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

Deferral of the surveillance requirement does not involve any physical changes to the plant nor does it change the way the plant is operated.

Thus, it can be concluded that deferring the surveillance requirement to the refueling outage cannot create the possibility of a different kind of accident from any accident previously evaluated.

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

Deferral of the surveillance requirement does not involve any physical changes to the plant nor does it change the way the plant is operated. The self-test feature and the monthly functional testing will provide reasonable assurance that the SECs will remain operable during the few weeks of deferral to the refueling outage. Also the ability to detect a degraded condition in the SEC will not be affected during the deferral period.

Therefore, the plant's response to accident conditions during the period of deferral will not be affected.

Thus, it can be reasonably concluded that this proposal to amend the Salem Unit 1 Technical Specifications, on a one-time basis, to defer surveillance requirement 4.3.2.1.3 does not involve a significant reduction in any margin of safety.

4.8.2.3.2.f, (Electrical Power Systems, 125 Volt D.C. Distribution), and 4.8.2.5.2.c.2 and 4.8.2.5.2.d (Electrical Power Systems, 28 Volt D.C. Distribution)

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

The deferral of the battery service tests to the refueling outage does not involve any physical changes to the power plant or to the manner in which the power plant is operated. Therefore, the probability of an accident previously evaluated is not increased.

Weekly and quarterly testing and performance monitoring by the system manager along with the current condition of the batteries (past test results demonstrating above 100% capacity) provide assurance that battery condition and performance will not deteriorate during the deferral period. Other positive industry experience for similar batteries on 24 month cycles also support this assurance. Therefore, the consequences of a loss of power accident will not be increased due to the deferral of the surveillance requirements.

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

The deferral of the battery service tests to the refueling outage does not involve any physical changes to the power plant or to the manner in which the power plant is operated. No new failure mechanisms will be introduced by the surveillance deferral. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The deferral of the battery service tests to the refueling outage does not involve any physical changes to the power plant or to the manner in which the power plant is operated. Continuing weekly and quarterly testing and performance monitoring along with the current condition of the batteries provides assurance that battery condition and performance will be acceptable during the deferral period and that any degradation that may occur will be detected. Therefore, the plant's response to accident conditions during the period of deferral will not be affected.

Thus, it can be reasonably concluded that this proposal to amend the Salem Unit 1 Technical Specifications, on a one-time basis, to defer surveillance requirements 4.8.2.3.2.f, 4.8.2.5.2.c.2 and 4.8.2.5.2.d does not involve a significant reduction in any margin of safety.

4.8.3.1.a.l.a, 4.8.3.1.a.l.b (Electric Power Systems, Electrical Equipment Protective Devices)

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

The deferral of inspection, calibration and meggering of 1A, 1B, 1C 460VAC transformer relays and current transformers (CT's); and inspection, calibration and meggering of 1F 4KV Bus Overload Relays to the refueling outage does not involve any physical changes to the power plant or to the manner in which the power plant is operated. Therefore, the probability of an accident previously evaluated is not increased.

The condition of the equipment as found for the three most recent completed surveillances (i.e. no failures or equipment problems found, no repair actions required, and test results satisfactory in all cases) provides assurance that equipment condition and performance will be acceptable during the deferral period. The subject equipment has performed well over the past several years and has demonstrated satisfactory stability and reliability. The plant's response to accident conditions during the period of deferral will not be affected. Therefore, the consequences of an accident previously evaluated will not be increased due to the deferral of the surveillance requirements.

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

The deferral of inspection, calibration and meggering of 1A, 1B, 1C 460VAC transformer relays and current transformers (CT's); and inspection, calibration and meggering of 1F 4KV Bus Overload Relays to the refueling outage does not involve any physical changes to the power plant or to the manner in which the power plant is operated. No new failure mechanisms will be introduced by the surveillance deferral. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The deferral of inspection, calibration and meggering of 1A, 1B, 1C 460VAC transformer relays and current transformers (CT's); and inspection, calibration and meggering of 1F 4KV Bus Overload Relays to the refueling outage does not involve any physical changes to the power plant or to the manner in which the power plant is operated. The results of previous tests which demonstrate the reliable and stable operation of the equipment over recent years provides assurance that the equipment will operate as designed during the deferral period. The plant's response to accident conditions during the period of deferral will not be affected.

Thus, it can be reasonably concluded that this proposal to amend the Salem Unit 1 Technical Specifications, on a one-time basis, to defer surveillance requirements 4.8.3.1.a.l.a and 4.8.3.1.a.l.b does not involve a significant reduction in any margin of safety.

4.1.2.2.c (Reactivity Control Systems, Flow Paths—Operating), 4.3.1.1.1, Table 4.3–1 (Reactor Trip System Instrumentation—Surveillance Requirements); 4.3.2.1.1, Table 4.3–2 (Engineered Safety Feature Actuation System Instrumentation—Surveillance Requirements); 4.5.1.d (Emergency Core Cooling Systems, Accumulators); 4.5.2.e.1 (Emergency Core Cooling Systems, ECCS Subsystems—Tave [greater than or equal to] 350 °F); 4.7.6.1.d.2 (Plan Systems, Control Room Emergency Air Conditioning System); and 4.7.10.b (Plant Systems, Chilled Water System—Auxiliary Building Subsystem)

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

The deferral of the Manual Safety Injection (SI) surveillance test to the refueling outage does not involve any physical changes to the power plant or to the manner in which the power plant is operated. Therefore, the probability of an accident previously evaluated is not increased.

Other surveillance testing provides assurance that the equipment will be reliable during the short deferral period. This testing, in conjunction with successful previous SI test results assure that the equipment will function properly during the short deferral period. Therefore, the consequences of an accident previously evaluated will not be increased due to the deferral of the surveillance requirements.

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

The deferral of the Manual Safety Injection (SI) surveillance test to the refueling outage does not involve any physical changes to the power plant or to the manner in which the power plant is operated. No new failure mechanisms will be introduced by the surveillance deferral. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The deferral of the Manual Safety Injection (SI) surveillance test to the refueling outage does not involve any physical changes to the power plant or to the manner in which the power plant is operated. Other surveillance testing in conjunction with successful previous SI test results provides assurance that the equipment will be reliable during the short deferral period. The plant's response to accident conditions during the period of deferral will not be affected.

Thus, it can be concluded that this proposal to amend the Salem Unit 1 Technical Specifications, on a one-time basis, to defer surveillance requirements 4.1.2.2.c; 4.3.1.1.1, Table 4.3–1; 4.3.2.1.1, Table 4.3–2;

4.5.1.d; 4.5.2.e.1; 4.7.6.1.d.2; and 4.7.10.b does not involve a significant reduction in any margin of safety.

4.8.1.1.2.d.7 (Electrical Power Systems, A.C. Power Sources) Diesel Generator 24 Hour Endurance Run)

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

Deferral of performance of the diesel generator 24 hour endurance runs to 1R13 does not involve any physical changes to the power plant or to the manner in which the power plant is operated. Therefore, the probability of an accident previously evaluated is not increased.

Based of the favorable history for previous endurance runs for the six Sale Unit 1 & 2 emergency diesel generators, continued normal monthly surveillance testing and the trending of engine and generator parameters, diesel generator operability can be assured during the deferral period. Therefore, the consequences of an accident previously evaluated will not be increased due to the deferral of the surveillance requirements.

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

Deferral of performance of the diesel generator 24 hour endurance runs to 1R13. does not involve any physical changes to the power plant or to the manner in which the power plant is operated. No new failure mechanisms will be introduced by the surveillance deferral. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Deferral of performance of the diesel generator 24 hour endurance runs to 1R13 does not involve any physical changes to the power plant or other manner in which the power plant is operated. Satisfactory endurance run history, other surveillance testing and performance monitoring assures diesel generator operability during the deferral period.

The plant's response to accident conditions during the period of deferral will not be affected.

Thus, it can be conducted that this proposal to amend the Salem Unit 1 Technical Specifications, on a one-time basis, to defer surveillance requirement 4.8.1.1.2.d.7 does not involve a significant reduction in any margin of safety.

4.3.1.1.1, Table 4.3–1 (Reactor Trip System Instrumentation-Surveillance Requirements); 4.3.3.5, Table 4.3–6 (Remote Shutdown Monitoring Instrumentation Surveillance Requirements); 4.3.3.7, Table 4.3–11 (Surveillance Requirements for Accident Monitoring Instrumentation)

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

Deferral of calibration of Pressurizer Level Channel 1, and the Containment Sump Level devices to 1R13 does not involve any physical changes to the power plant or to the manner in which the power plant is operated. Therefore, the probability of an accident previously evaluated is not increased.

Review of trends of the level channels during the current operating cycle and continued monitoring of the channels provides reasonable assurance that the channels will perform their design function during the deferral period. Therefore, the consequences of an accident previously evaluated will not be increased due to the deferral of the surveillance requirements.

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

Deferral of calibration of Pressurizer Level Channel 1, and the Containment Sump Level devices to 1R13 does not involve any physical changes to the power plant or to the manner in which the power plant is operated. No new failure mechanisms will be introduced by the surveillance deferral. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Deferral of calibration of Pressurizer Level Channel 1, and the Containment Sump Level devices to 1R13 does not involve any physical changes to the power plant or to the manner in which the power plant is operated. Review of trends of the level channels during the current operating cycle and continued monitoring provides reasonable assurance that the channels will perform their design function during the deferral period. There will be no effect on the response to accident conditions during the period of deferral.

Thus, it can be concluded that this proposal to amend the Salem Unit 1 Technical Specifications, one a one-time basis, to defer surveillance requirements 4.3.1.1.1, Table 4.3–1, item 11; 4.3.3.5, Table 4.3–6, item 2; and 4.3.3.7, Table 4.3–11, items 4 and 17 does not involve a significant reduction in any margin of safety.

Administrative and Editorial Change

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

The proposed changes are administrative or editorial and do not involve any physical changes to the plant. The administrative changes and editorial changes do not delete any existing surveillance requirements or delete any requirements from the Limiting Condition for Operations (LCOs) or Action Statements and therefore do not reduce the actions that are currently taken to demonstrate operability of plant structures, systems, or components (SSCs). The additional surveillance requirement that is being added including the new surveillance corrects a past administrative error and should have been incorporated within the Tech Specs as part of an approved Amendment. This change will provide additional assurance that SSCs perform their intended safety functions. Surveillance testing has been and is currently being performed for the surveillance requirement that should have been incorporated and is now administratively being added to the Tech Specs. Since these changes do not modify any SSCs or reduce the current requirements for demonstrating operability of these SSCs, the proposed changes to the Tech Specs do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes to the Tech Specs are administrative and editorial corrections that do not affect the ability of the plant systems to meet their current Tech Spec requirements or design basis functions. There is no reduction in the current surveillance requirements required to demonstrate the operability of plant SSCs. These changes also do not involve any physical changes to plant SSCs. Therefore the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes are administrative and editorial corrections that do not affect the ability of plant SSCs to perform their design basis accident functions. 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.

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, NJ 08079.

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

NRC Project Director: William M. Dean.

Southern California Edison Company, Et Al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego County, California

Date of amendment requests: September 4, 1998 as modified December 7, 1998.

Description of amendment requests: The proposed amendment would modify the Technical Specifications (TS) to increase the allowed as-found pressurizer safety valve setpoint tolerance from +/−1 percent to +3/−2 percent.

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.

All Updated Final Safety Analysis Report (UFSAR) Chapter 15 events have been evaluated to determine the impact of the increases in as found Pressurizer Safety Valve (PSV) tolerance from +1% and −1% to +3% and −2%. The events that result in challenging the opening of the PSVs are Loss of Condenser Vacuum With and Without Single Failure, Loss of Normal Feedwater Flow, Feedwater System Pipe Breaks, Total Loss of Reactor Coolant System (RCS) Flow, Uncontrolled Control Element Assembly (CEA) Withdrawal, CEA Ejection, Chemical and Volume Control System (CVCS) Malfunction With and Without Single Failure, Inadvertent Emergency Core Cooling System (ECCS) Actuation With and Without Single Failure, and Inadvertent Opening of a PSV. Of these, the limiting events are the Loss of Condenser Vacuum (LOCV), Loss of Condenser Vacuum With a Concurrent Single Failure of an Active Component (LOCVsf), CVCS Malfunction, CVCS Malfunction With a Concurrent Single Failure of an Active Component, and Feedwater System Pipe Breaks. These limiting events have been reanalyzed for the wider PSV tolerance. For all the reanalyzed events it is assumed that plant operation is maintained at a maximum pressurizer level of 57%. For the CVCS Malfunction With and Without Single Failure Events and the Inadvertent ECCS Actuation With and Without Single Failure Events, it is also assumed that the operator can respond within 15 minutes to mitigate the event.

The change in as found PSV tolerance from −1% to −2% results in the earlier opening of the PSVs for the analyzed events. To compensate for this earlier opening of the PSVs the high pressurizer pressure trip analysis setpoint was reduced from 2437 psia (non-harsh environment) and 2450 (harsh environment) to 2410 psia (non-harsh environment) and 2434 (harsh environment). These setpoint changes insure that the high pressurizer pressure trip is actuated sufficiently early before the opening of the PSVs such that no liquid is released through the PSVs. Therefore, the change to the PSV negative tolerance does not result in a significant increase in the probability or consequences of any previously evaluated accident.

The change in PSV as found tolerance from +1% to +3% results in later opening of the PSVs for the analyzed events. The PSV actuation to mitigate the consequences of the analyzed accidents are thus delayed. However, the lowering of the high pressurizer pressure trip setpoint, as discussed above, mitigates the increase in peak primary pressure and assures that no liquid is released through the PSVs. Therefore, this change to the PSV positive tolerance does not result in a significant increase in the probability or consequences of any previously analyzed design basis event.

There are no other changes to the plant equipment or operation which could create an increase in the probability or consequences of any event previously evaluated.

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

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

Operation in accordance with this proposed change will not involve any change to plant equipment or operation which could create a new or different kind of accident. The as-left PSV tolerance will continue to remain at +/−1%. The change in as-found tolerance of the PSVs to −2% and +3% will not introduce the possibility of a new or different kind of accident because evaluation of the design basis events shows that no water is expected to be released through the PSVs.

There are no other changes to the plant equipment or operation which could create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

Operation in accordance with this proposed change will not change the manner in which safety limits, limiting safety settings, or limiting conditions for operation are determined. The acceptance criteria for all of the events reanalyzed include an appropriate margin of safety.

There are no changes to the acceptance criteria nor are the acceptance criteria exceeded for these events assuming plant operation at a maximum pressurizer level of 57% and operator response time of 15 minutes for the CVCS Malfunction With and Without Single Failure Events and the Inadvertent ECCS Actuation With and Without Single Failure Events.

Therefore, this 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Local Public Document Room location: Main Library, University of California, P.O. Box 19557, Irvine, California 92713.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, P.O. Box 800, Rosemead, California 91770.

NRC Project Director: William H. Bateman.

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

Date of application for amendments: January 15, 1999 (TS 98–09).

Brief description of amendments: The proposed amendments would change the Sequoyah (SQN) Technical Specification (TS) requirements by relocating Section 3.3.3.3, ''Seismic Instrumentation,'' to the SQN Technical Requirements Manual (TRM).

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

TVA has concluded that operation of SQN Units 1 and 2, in accordance with the proposed change to the TS, does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

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

A. The proposed revision to the TS relocates the requirements for SQN seismic instrumentation without changing the current requirements. TVA does not consider the instrumentation to be the source of any accident; therefore, this administrative relocation of the requirements will not increase the possibility of an accident. The capability of the seismic instrumentation will continue to provide the same function of data collection. Changes to the relocated requirements will be processed, in accordance with 10 CFR 50.59, to ensure the seismic instrumentation functions will be properly maintained. Therefore, the proposed relocation of the seismic instrumentation requirements will not increase the consequences of an accident.

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

The SQN seismic instrumentation is used to record data for use in evaluating the effect of a seismic event. This instrumentation is not associated with accident mitigation or previously evaluated accidents and would not be the initiator of any new or different kind of accident. The proposed change does not alter the current functions of SQN's seismic instrumentation; therefore, this proposed change will not create the possibility of a new or different kind of accident.

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

The requirements for SQN's seismic instrumentation are unchanged by the proposed relocation of the requirements to the SQN TRM. The function of the seismic instrumentation and SRs to ensure operability of the instrumentation remains unchanged. Any future changes to these requirements will be evaluated, in accordance with 10 CFR 50.59, to ensure acceptability and NRC review as required. Accordingly, the proposed change will not result in a reduction in a margin of safety.

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

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

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

NRC Project Director: Cecil O. Thomas.

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

Date of amendment request: April 23, 1998.

Description of amendment request: The amendment request proposes changes to the existing requirements for the RHR Service Water (RHRSW), Station Service Water (SSW) and Alternate Cooling Tower Systems (ACS) as identified in Technical Specifications (TS) 4.5.C and 3/4.5.D.

Specifically, the changes proposed are as follows:

(1) Specifications 3.5.D.3 and 4.5.D.3: This requirement is revised to delete the existing allowance for 7 days of operation after both SSW subsystems are made or found to be inoperable.

(2) Specification 4.5.C.1 and Specification 4.5.D.1: These requirements have been revised to relocate testing information related to pump flow and pressure testing characteristics for the RHRSW and SSW Systems, respectively, to the TRM.

(3) Specifications 3.5.D.1, 3.5.D.2, 3.5.D.3, 4.5.D.2, 4.5.D.3 and associated Bases: All reference to SSW ''subsystem'' has been replaced by ''essential equipment cooling loop'' to more accurately reflect VYNPS design and operation. In addition, certain operability clarifications have been made to the Bases relative to affected Specifications.

(4) Bases for Specifications 3.5.D: The Bases have been revised to omit statements which imply that the ACS could provide adequate heat removal following a postulated accident. Other Bases additions have been made which include certain operability clarifications relative to affected 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:

For change No. 1:

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

This change deletes the existing allowance for 7 days of operation after both Station Service Water (SSW) subsystems are made or found to be inoperable. At least one subsystem of the SSW System is required to be operable to mitigate the consequences of a design basis accident. Therefore, with both subsystems inoperable, the unit is required to shut down. Current Technical Specifications (TS) erroneously allow 7 days of operation after both SSW subsystems are made or found to be inoperable before requiring that the reactor be placed in cold shutdown within 24 hours. This allowance is incorrectly based on the assumption that the Alternate Cooling Tower System (ACS) is able to fulfill the post-accident heat removal requirements when both SSW Subsystems are made or found to be inoperable. Since the ACS is not capable of fulfilling this backup role, the allowance for seven days of operation with both SSW Subsystems inoperable is removed, and a requirement to shutdown the unit is provided in its place. This proposed change deletes the allowance for 7 days of operation in this condition, and instead requires an orderly shutdown to be initiated and the reactor to be placed in cold shutdown within 24 hours. Since the same amount of time is allowed to conduct the required shutdown, this change will not significantly increase the consequences of any previously analyzed accident. In addition, the SSW system is not considered to be the initiator of any previously analyzed accident. Therefore, this change will not significantly increase the probability or consequences of any previously analyzed accident.

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

This change will not physically alter the plant (no new or different types of equipment will be installed). The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions. Therefore, this change will 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?

This change deletes the existing allowance for 7 days of operation after both SSW subsystems are made or found to be inoperable. At least one subsystem of the SSW System is required to be operable to mitigate the consequences of a design basis accident. Therefore, with both subsystems inoperable, the unit is required to be shut down. Current TS requirements erroneously allow 7 days of operation after both the SSW subsystems are made or found to be inoperable before requiring that the reactor be placed in cold shutdown within 24 hours. This allowance is incorrectly based on the assumption that the ACS is able to fulfill the post-accident heat removal requirements when both SSW Subsystems are inoperable. Since the ACS is not capable of fulfilling this backup role, the allowance for seven days of operation with both SSW Subsystems inoperable is removed, and a requirement to shutdown the unit within 24 hours is provided in its place. Therefore, elimination of the allowance for 7 days of operation with both SSW Subsystems inoperable does not involve a significant reduction in a margin of safety.

For change No. 2:

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

The proposed change relocates testing information details for the Residual Heat Removal Service Water (RHRSW) and Station Service Water (SSW) systems, respectively, to the Technical Requirements Manual (TRM) under the control of 10 CFR 50.59. These controls are adequate to ensure the required testing is performed to verify operability. As such, these relocated details are not required to be in the Technical Specifications to provide adequate protection of the public health and safety. Changes to these relocated requirements in the TRM will be controlled by 10 CFR 50.59. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose or eliminate any requirements and adequate control of the information will be maintained. Thus, this 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 proposed change will not reduce a margin of safety because the simple relocation of testing details from the TS to the TRM has no impact on any safety analyses assumptions. Since any future changes to these requirements will be evaluated per the requirements of 10 CFR 50.59, no reduction in a margin of safety will be allowed. Therefore, this change does not involve a significant reduction in the margin of safety.

For change No. 3:

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

This change proposes to revise the wording of Station Service Water (SSW) Specifications to replace ''subsystem'' with ''essential equipment cooling loop'' to more accurately reflect VYNPS design and operation. At least two SSW pumps and one essential equipment cooling loop of the SSW System are required to be operable to mitigate the consequences of a design basis accident. Since this proposed change represents no change to existing requirements, this change will not significantly increase the consequences of any previously analyzed accident. In addition, SSW is not considered to be the initiator of any previously analyzed accident. Therefore, this change will not significantly increase the probability or consequences of any previously analyzed accident.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose or eliminate any requirements and adequate control of existing requirements will be maintained. Thus, this 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 proposed change continues to provide the previous margin of safety regarding the capability to remove post-accident heat loads. At least two SSW pumps and one essential equipment cooling loop will be required to be operable or the unit will be required to be shutdown within 24 hours. Since this is the same basis both before and after the change, this change 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 amendment request involves no significant hazards consideration.

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

Attorney for licensee: Mr. David R. Lewis, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

NRC Project Director: William M. Dean.

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

Date of amendment request: November 2, 1998.

Description of amendment request: The licensee proposed to modify the Technical Specifications to more clearly describe the Emergency Core Cooling System Actuation Instrumentation—Low Pressure Coolant Injection (LPCI) System A/B Residual Heat Removal (RHR) Pump Start time delay requirements and the Core Spray System A/B Pump Start time delay requirements.

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

Change #1: Deletion of the O second time delay for first RHR pump (A/D) start.

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

The proposed change does not involve a change to the plant design or operation. The instantaneous relays installed under corrective actions of LER 96–027 were evaluated as being equivalent in meeting the plant design of a 0 second time delay (instantaneous start) and an improvement on the minimum 500 millisecond time delay relays previously installed. The intent is to get LPCI flow started as soon as possible within the limits of the emergency bus power supply. The instantaneous start provides for a faster flow initiation. The proposed change does not affect any of the parameters or conditions that contribute to initiation of any accidents previously evaluated. Therefore, the proposed change cannot increase the probability of an accident previously evaluated.

The proposed change does not involve a change in the operation of the relay controlling the initial RHR pump start on a [loss of coolant accident] LOCA with normal AC power not available. The instantaneous logic sequence relay functions to start the initial RHR Pump within 35 milliseconds of re-energization of the associated Emergency Bus. This start time is consistent with the plant safety analysis and [emergency diesel generator] EDG load analysis, therefore, the proposed change does not significantly increase the consequences of any accident previously evaluated.

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

This proposed change will not involve any physical changes to plant structures, systems or components (SSC), or the manner in which these SSCs are operated or maintained. Deletion of the 0 second Time Delay Trip Function and associated calibration requirement will not affect initial RHR pump starting on a LOCA signal with normal AC power not available. The instantaneous logic sequence relay will still be tested under the Trip System Logic Functional Test at a frequency of once per operating cycle. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This proposed change to delete the 0 second Time Delay Trip Function and associated calibration requirement will not change operation of the initial RHR Pump start on a LOCA signal with normal power not available. The instantaneous logic sequence relay will function to initiate RHR Pump A/D start within 35 milliseconds of re-energization of the associated Emergency Bus, therefore, water will be delivered as designed. This RHR Pump start time is within the assumptions of the LOCA safety analysis of record. Therefore, this change does not involve a significant reduction in a margin of safety.

Change #2: Addition of a 3 second lower limit to the trip level setting for the second RHR pump (B/C) start time delay trip function.

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

The proposed change does not involve a change to the plant design or operation. The proposed change is more restrictive than existing Technical Specifications for this function. The proposed change limits the low value Trip Level Setting of the time delay relay and thus provides for EDG recovery from the initial RHR Pump (A/D) start. As a result, the proposed change does not affect any of the parameters or conditions that contribute to initiation of any accidents previously evaluated. The equipment will still start within the assumptions of the LOCA safety analysis of record. Thus, the proposed change cannot increase the probability of an accident previously evaluated.

The proposed change ensures that the EDG has sufficient time to recover from the loading of the first RHR pump (A/D) prior to the loading of the second RHR pump (B/C). This load sequencing is experienced during a LOCA with normal AC power not available, thus providing increased reliability. Therefore, the proposed change will not result in a significant change in the consequences of any accident previously evaluated.

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

This proposed change will not involve any physical changes to plant systems, structures or components (SSC), or the manner in which these SSCs are intended to be operated or maintained. Addition of the 3 second lower limit on the second RHR Pump (B/C) Start Time Delay Function will ensure that, on a LOCA signal with normal AC power not available, the EDG voltage and frequency will adequately recover prior to the second RHR pump start. The instantaneous logic sequence relay will still be tested under the Trip System Logic Functional Test each Operating Cycle. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This proposed change to include a 3 second lower limit to the second RHR Pump Start Time Delay Trip Function will not change operation of the second RHR Pump start on a LOCA signal (without normal power available). The proposed change will ensure sufficient time is available for the EDG to recover from the initial RHR Pump (A/D) start. The proposed second RHR Pump Start Time Delay Trip Level Setting of 3 [less than or equal to] t [less than or equal to] 5 seconds is within the assumptions of the LOCA evaluation and analysis of FSAR Sections 6.5 and 8.5. Therefore, this change does not involve a significant reduction in a margin of safety.

Change #3: Addition of an 8 second lower limit to the trip level setting for the core spray pump (A/B) start time delay trip function.

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

The proposed change does not involve a change to the plant design or operation. The proposed change is more restrictive than existing Technical Specifications for this function. The proposed change limits the low value Trip Level Setting of the time delay relay and thus provides for EDG recovery following the RHR B/C Pump start. As a result, the proposed change does not affect any of the parameters or conditions that contribute to initiation of any accidents previously evaluated. The equipment will still start within the assumptions of the LOCA analysis of record. Thus, the proposed change cannot increase the probability of an accident previously evaluated.

The proposed change ensures that the EDG has sufficient time to recover following the loading of the B/C RHR pump and prior to the loading of the associated Core Spray pump. This load sequencing is experienced during a LOCA without normal power available, thus providing increased reliability. Therefore, the proposed change will not result in a significant change in the consequences of any accident previously evaluated.

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

This proposed change will not involve any physical changes to plant structures, systems or components (SSC), or the manner in which these SSCs are intended to be operated or maintained. Addition of the 8 second lower limit on the Core Spray Pump Start Time Delay Trip Function will ensure that, on a LOCA signal (with normal power not available) the EDG voltage and frequency will adequately recover prior to the Core Spray pump start. The Core Spray instantaneous logic sequence relays (normal AC available) and the CS Pump Start Time Delay relays will still be tested under the Trip System Logic Functional Test each Operating Cycle. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This proposed change to include an 8 second lower limit to the Core Spray Pump Start Time Delay Trip Function will not change operation of the Core Spray Pump start on a LOCA signal with normal AC power not available. The proposed change will ensure sufficient time is available for the EDG to recover from the previous RHR Pump start. The proposed Core Spray Pump Start Time Delay Trip Level Setting of 8 [less than or equal to] t [less than or equal to] 10 seconds is within the assumptions of the LOCA evaluation and analysis of FSAR Sections 6.5 and 8.5. Therefore, this 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.

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

Attorney for licensee: Mr. David R. Lewis, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

NRC Project Director: William M. Dean.

Virginia Electric and Power Company, Docket Nos. 50–280 and 50–281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia

Date of amendment request: September 24, 1998.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) to modify the testing requirements for the reactor trip bypass breakers. The current TS require the bypass breakers to be tested ''prior to being placed in service.'' The proposed changes will allow the bypass breakers to be tested immediately after placing the breaker in service, but prior to commencing Reactor Protection System testing or maintenance.

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:

(a) Operation and testing of the reactor trip bypass breakers does not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.

The testing sequence will continue to ensure that the reactor trip system will be operable to mitigate the consequences of any unsafe or improper reactor operation during steady state or transient power operations. During the short period of time the breaker is closed before the undervoltage trip device test, the operability of the breaker is established based on satisfactory breaker testing conducted during the previous surveillance interval. Although the breaker is placed in service before it is tested, the breaker is tested as soon as practicable to verify operability prior to performing testing of the reactor trip system or required maintenance. Therefore, the proposed test sequence does not significantly increase the probability of occurrence or the consequences of any previously analyzed accident.

(b) The proposed Technical Specifications do not create the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report.

The proposed test sequence change does not alter the actual test performed to establish operability of the reactor trip bypass breakers. The bypass breakers will be proven operable prior to reactor trip system testing or required maintenance. During the short period of time the breaker is closed before the undervoltage trip device test, the operability of the breaker is established based on satisfactory breaker testing conducted during the previous surveillance interval. Although the breaker is placed in service before it is tested, the breaker is tested as soon as practicable to verify operability prior to performing testing of the reactor trip system or required maintenance. Therefore, it is concluded that no new or different kind of accident or malfunction from any previously evaluated has been created.

(c) The proposed Technical Specifications change does not result in a significant reduction in margin of safety.

The proposed change in the reactor trip bypass breaker test sequence provides assurance that the reactor trip system remains operable during normal operations or during reactor trip system testing and required maintenance to mitigate the consequences of any unsafe or improper reactor operation. Therefore, the proposed change in the test sequence for the reactor trip bypass breaker does not significantly reduce the margin of safety.

This analysis demonstrate that the proposed amendment to the Surry Units 1 and 2 Technical Specifications does not involve a significant increase in the probability or consequences of a previously evaluated accident, does not create the possibility of a new or different kind of accident and does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Swem Library, College of William and Mary, Williamsburg. Virginia 23185.

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

NRC Project Director: Herbert N. Berkow.

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.

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

Date of amendment request: December 7, 1998.

Description of amendment request: The proposed amendments would correct the lube oil inventory requirement from a range of 575–600 gallons to a range of 375–400 gallons.

Date of publication of individual notice in ***Federal Register***: December 2, 1998 (63 FR 66591).

Expiration date of individual notice: January 4, 1999.

Local Public Document Room location: York County Library, 138 East Street, Rock Hill, South Carolina.

Northeast Nuclear Energy Company (NNECO), Et Al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: January 4, 1999.

Description of amendment request: The proposed amendment would change Technical Specifications (TSs) 3.5.2, ''Emergency Core Cooling Systems—ECCS Subsystems-Tavg [greater than or equal to] 300 [degrees Fahrenheit];'' 3.6.2.1, ''Containment Systems—Depressurization and Cooling Systems—Containment Spray and Cooling Systems;'' 3.7.1.2, ''Plant Systems—Auxiliary Feedwater Pumps;'' 3.7.3.1, ''Plant Systems—Reactor Building Closed Cooling Water System;'' and 3.7.4.1, ''Plant Systems—Service Water System.'' Changes to the acceptance criteria contained in these TSs are necessary based on revised hydraulic analyses and related accident analyses. Also, the bases of the associated TSs will be modified to address the proposed changes.

Date of publication of individual notice in **Federal Register:** January 14, 1999 (64 FR 2523).

Expiration date of individual notice: February 16, 1999.

Local Public Document Room: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, or the Waterford Public Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

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 at the local public document rooms for the particular facilities involved.

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

Date of application for amendments: December 7, 1998.

Brief description of amendments: The amendments revise Technical Specification Section 3.8.3 to correct the lube oil inventory requirement from a range of 575–600 gallons to a range of 375–400 gallons.

Date of issuance: January 15, 1999.

Effective date: As of the date of issuance to be implemented concurrently with implementation of Amendment Nos. 173 (Unit 1) and 165 (Unit 2).

Amendment Nos.: 175—Unit 1; 167—Unit 2.

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

Date of initial notice in **Federal Register:** December 16, 1998 (63 FR 69328).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 15, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina.

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

Date of amendment request: June 29, 1998.

Brief description of amendment: The amendment revises the Applicability of Technical Specification (TS) 3.4.2, ''Reactor Coolant System—Safety Valves—Shutdown.'' An associated action is also revised and a footnote is removed. The amendment also revises TS 3.4.12, ''Reactor Coolant System—Overpressure Protection,'' allowing safety injection tanks to remain unisolated if they are pressurized to less than 300 psig and making some editorial changes. In addition, affected index and Bases pages are revised.

Date of issuance: January 19, 1999.

Effective date: The license amendment is effective as of its date of issuance with full implementation within 60 days.

Amendment No.: 199.

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

Date of initial notice in ***Federal Register:*** October 21, 1998, (63 FR 56243).

The Commission's related evaluation of the amendments is contained in a

Safety Evaluation dated January 19, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 29, 1998.

Brief description of amendment: The amendment approves a change to the Technical Specifications (TS) Table 3.3–4, ''Engineered Safety Feature Actuation System Instrumentation,'' to provide a range of acceptable values for the 4 KV buss loss of voltage relays rather than a single value as currently recorded in the TS. In addition minor changes were made to the trip time delay.

Date of issuance: January 26, 1999.

Effective date: The license amendment is effective as of its date of issuance and shall be implemented prior to the facility's restart from refueling outage 2R13.

Amendment No.: 200.

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

Date of initial notice in ***Federal Register:*** October 21, 1998 (63 FR 56244).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 26, 1999.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: September 22, 1998.

Brief description of amendment: The amendment deleted license conditions associated with the River Bend Station (RBS) Transamerica Delaval, Inc. (TDI) emergency diesel generators (EDGs), which prescribed various inspection requirements following an EDG overload condition. The License Conditions were originally issued following the publication of NUREG 1216, which called for extensive periodic engine tear-downs as the major part of a maintenance and surveillance program for TDI engines. The removal of the aforementioned license conditions is consistent with the NRC's approval of Generic Topical Report TDI–EDG–001–A ''Basis for Modification to Inspection Requirements for Transamerica Delaval, Inc., Emergency Diesel Generators''. EOI will continue to inspect and maintains its EDGs in accordance with Technical Requirements Manual (TRM) surveillance requirement TSR 3.8.1.21. Periodicity of planned inspections and maintenance are based upon the manufacturer's recommendations for standby service.

Date of issuance: January 27, 1999.

Effective date: January 27, 1999.

Amendment No.: 102.

Facility Operating License No. NPF–47: The amendment revised the operating license.

Date of initial notice in ***Federal Register:*** November 4, 1998 (63 FR 59592).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 27, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803.

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

Date of application for amendment: August 24, 1998, as supplemented November 20, 1998.

Brief description of amendment: The amendment approves operator action for meeting the ''ready-to-load'' requirement for the Division 3 diesel generator.

Date of issuance: January 19, 1999.

Effective date: January 19, 1999.

Amendment No.: 119.

Facility Operating License No. NPF–62: The amendment authorized revision of the Updated Safety Analysis Report.

Date of initial notice in ***Federal Register.*** September 10, 1998 (63 FR 48529).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 19, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: The Vespasian Warner Public Library, 120 West Johnson Street, Clinton, IL 61727.

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

Date of application for amendment: July 31, 1998.

Brief description of amendment: The amendment clarifies requirements for diesel generator start voltage and frequency.

Date of issuance: January 20, 1999.

Effective date: January 20, 1999.

Amendment No.: 120.

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

Date of initial notice in ***Federal Register:*** October 7, 1998 (63 FR 53949).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 20, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: The Vespasian Warner Public Library, 120 West Johnson Street, Clinton, IL 61727.

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

Date of amendment request: March 2, 1998.

Description of amendment request: The amendment changes the Technical Specifications by eliminating the emergency diesel generator accelerated testing and special reporting requirements of TS 4.8.1.1.2a, 4.8.1.1.3, Table 4.8–1 and 4.8.1.2 in accordance with Generic Letter 94–01.

Date of issuance: January 21, 1999.

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

Amendment No.: 59.

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

Date of initial notice in ***Federal Register:*** April 22, 1998 (63 FR 19971). The Commission received comments which were addressed in the staff's Safety Evaluation dated January 21, 1999.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 21, 1999.

No significant hazards consideration comments received: Yes.

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

Northeast Nuclear Energy Company (NNECO), Et Al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of application for amendment: September 28, 1998.

Brief description of amendment: The amendment approves the previously implemented revision to the Final Safety Analysis Report (FSAR) Section 8.7.3.1 that changed certain electrical separation requirements from 12 inches to 6 inches. The FSAR change was previously implemented following an erroneous 10 CFR 50.59 evaluation.

Date of issuance: January 20, 1999.

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

Amendment No.: 224.

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

Date of initial notice in **Federal Register**: November 4, 1998 (63 FR 59593).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 20, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

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

Date of application for amendment: November 10, 1998.

Brief description of amendment: The amendment changes Technical Specifications 3.3.1.1, ''Reactor Protective Instrumentation,'' and 3.3.2.1, ''Engineered Safety Feature Actuation System Instrumentation,'' to restrict the time a reactor protection or engineered safety feature actuation channel can be in the bypass position for 48 hours, from an indefinite period of time.

Date of issuance: January 27, 1999.

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

Amendment No.: 225.

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

Date of initial notice in **Federal Register**: December 16, 1998 (63 FR 69343).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 27, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

PECO Energy Company, Docket No. 50–353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Date of application for amendment: September 14, 1998.

Brief description of amendment: This amendment revised Limerick Generating Station, Unit 2, Technical Specification (TS) Table 4.4.6.1.3–1, ''Reactor Vessel Material Surveillance Program—Withdrawal Schedule.'' The revision changed the schedule for withdrawing the first surveillance capsule from 8 Effective Full Power Years (EFPY) to 15 EFPY, and the second surveillance capsule from 20 EFPY to 30 EFPY. A revision to the TS Surveillance Requirement (SR) has also been made. This revision removed the reference to flux wire removal and analysis that was originally required following the first cycle of operation. TS SR 4.4.6.1.4 was changed to refer to the flux wires that are located within the surveillance capsules, which will be removed and analyzed in accordance with the surveillance capsule removal schedule located in Table 4.4.6.1.3–1.

Date of issuance: January 12, 1999.

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

Amendment No.: 94.

Facility Operating License No. NPF–85: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** October 21, 1998 (63 FR 56253).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 12, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: October 15, 1998.

Brief description of amendments: These amendments add a new Technical Specification (TS) section and TS Bases section to incorporate a special test exception to allow reactor coolant temperatures greater than 200 °F but less than or equal to 212 °F during inservice testing and hydrostatic testing.

Date of issuance: January 12, 1999.

Effective date: Both units, as of the date of issuance, to be implemented within 30 days.

Amendment Nos.: 133 and 95.

Facility Operating License Nos. NPF–39 and NPF–85: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 18, 1998 (63 FR 64120).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 12, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: May 10, 1996, as supplemented on March 19 and August 29, 1997.

Brief description of amendments: This amendment incorporates into the Technical Specifications the Margin Recovery portion of the Fuel Upgrade Margin Recovery Program and supports increased steam generator plugging, improved fuel reliability, reduced fuel costs, longer fuel cycles, reduced spent fuel storage, and enhanced reactor safety. In a letter dated November 26, 1997, the Commission issued the amendment for Salem Unit 1.

Date of issuance: January 8, 1999.

Effective Date: January 8, 1999.

Amendment Nos.: 197.

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

Date of initial notice in ***Federal Register:*** July 3, 1996 (61 FR 34898).

The March 19, and August 29, 1997, letters 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 January 8, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, NJ 08079.

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

Date of application for amendment: July 1, 1998.

Brief description of amendment: The amendment revises Virgil C. Summer Nuclear Station Technical Specification Surveillance Requirement (SR) 4.7.7.e to remove the ''during shutdown'' condition from the specified test interval. The amendment also makes administrative changes to SR 4.7.7.g, and BASES 3/4.2.2 and 3/4.2.3 to correct typographical errors.

Date of issuance: January 27, 1999.

Effective date: January 27, 1999.

Amendment No.: 141.

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

Date of initial notice in ***Federal Register:*** October 7, 1998 (63 FR 53955).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 27, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180.

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 application for amendments: September 3, 1998, as supplemented by letter dated December 8, 1998.

Brief Description of amendments: The amendments change the Vogtle Electric Generating Plant, Units 1 and 2 Technical Specifications to: (1) Support the replacement of the Nuclear Instrumentation System Source Range and Intermediate Range Channels and Post-Accident Neutron Flux Monitoring System, and (2) delete the requirement for performing response time testing of the source range channels and power range detector plateau voltage determinations.

Date of issuance: January 22, 1999.

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

Amendment Nos.: Unit 1—104; Unit 2—82.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revise the Technical Specifications.

Date of initial notice in ***Federal Register***: October 7, 1998 (63 FR 53957).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 22, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Burke County Library, 412 Fourth Street, Waynesboro, Georgia.

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

Date of amendment request: October 29, 1998.

Brief description of amendments: Relocates the Technical Specification 3/4.3.4 requirements for Turbine Overspeed Protection to the Technical Requirements Manual.

Date of issuance: January 21, 1999.

Effective date: The license amendment is effective as of its date of issuance, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 1—Amendment No. 101; Unit 2—Amendment No. 88.

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

Date of initial notice in ***Federal Register***: December 16, 1998, (63 FR 69347). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 21, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

Tennessee Valley Authority, Docket No. 50–260, Browns Ferry Nuclear Plant, Unit No. 2, and Docket No. 50–296, Browns Ferry Nuclear Plant, Unit No. 3, Limestone County, Alabama

Date of amendment request: March 3, 1998 as supplemented November 13, and December 15, 1998.

Description of amendment request: The amendments revise the pressure-temperature limit curves in the Technical Specifications (TS) for BFN Units 2 and 3 to 16 and 20 effective full power years, respectively.

Date of issuance: January 15, 1999.

Effective date: January 15, 1999.

Amendment Nos.: 257 and 217.

Facility Operating License Nos. DPR–52 and DPR–68: Amendments revised the TS.

Date of initial notice in ***Federal Register***: April 22, 1998 (63 FR 19979). The licensee's letters of November 13, and December 15, 1998, did not expand the scope of the application or affect the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 15, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Athens Public Library, 405 E. South Street, Athens, Alabama 35611.

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

Date of application for amendment: June 26, 1998, as supplemented November 6, 1998. (TS 98–06).

Brief description of amendment: The amendment authorizes the deletion of the power range neutron flux high negative rate reactor trip function based on the analysis provided in Westinghouse Electric Corporation WCAP–11394–A, ''Methodology for the Analysis of the Dropped Rod Event.''

Date of issuance: January 15, 1999.

Effective date: January 15, 1999.

Amendment No.: 18.

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

Date of initial notice in ***Federal Register***: July 29, 1998 (63 FR 40562). The November 5, 1998, letter contained clarifying information that did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 15, 1998.

No significant hazards consideration comments received: None.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, TN 37402.

TU Electric Company, 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 11, 1998.

Brief description of amendments: Revises core safety limit curve and Overtemperature N–16 reactor trip setpoints based on analysis of the core configuration and expected operation for the CPSES Unit 2, Cycle 5. The changes apply equally to CPSES Units 1 and 2 licenses since the Technical Specifications are combined.

Date of issuance: January 29, 1999.

Effective date: The license amendment is effective as of its date of issuance, to be implemented within 90 days of issuance.

Amendment Nos.: Unit 1—Amendment No. 63; Unit 2—Amendment No. 49.

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

Date of initial notice in ***Federal Register:*** December 30, 1998 (63 FR 71974).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 29, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, TX 76019.

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

Date of application for amendment: February 24, 1998, as supplemented by letters dated May 27, June 25, August 25, September 3, November 3, and December 4, 1998.

Brief description of amendment: The amendment revised the technical specifications to allow an increase in the Callaway Plant, Unit 1 spent fuel pool storage capacity from 1344 fuel assemblies to 2363 fuel assemblies. The amendment also revises the technical specifications to allow storage of an additional 279 fuel assemblies in the cask loading pit.

Date of issuance: January 19, 1999.

Effective date: January 19, 1999, to be fully implemented no later than December 31, 1999, except that the racks in the cask loading pit may be installed at a future time after the completion of the next refueling outage.

Amendment No.: 129.

Facility Operating License No. NPF–30: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: July 13, 1998 (63 FR 37598).

The June 25, August 25, September 3, November 3, and December 4, 1998, supplemental letters provided additional clarifying information that did not change the staff's original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 19, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Elmer Ellis Library, University of Missouri, Columbia Missouri 65201.

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

Date of application for amendment: November 3, 1998.

Brief description of amendment: The amendment makes administrative changes to the Technical Specifications to correct errors, add consistency within the Technical Specifications, and make nomenclature changes to support and enhance usability of the Technical Specifications.

Date of Issuance: January 5, 1999.

Effective date: January 5, 1999, to be implemented within 30 days.

Amendment No.: 164.

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

Date of initial notice in ***Federal Register***: December 2, 1998 (63 FR 66605).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated January 5, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

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

Date of application for amendment: December 11, 1998.

Brief description of amendment: The amendment revises the Technical Specifications to allow manual containment isolation valves to be opened intermittently under administrative controls.

Date of Issuance: January 19, 1999.

Effective date: January 19, 1999, to be implemented within 30 days.

Amendment No.: 165.

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

Date of initial notice in ***Federal Register***: December 18, 1998 (63 FR 70168).

The Commission's related evaluation of this amendments is contained in a Safety Evaluation dated January 19, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

Washington Public Power Supply System, Docket No. 50–397, Nuclear Project No. 2, Benton County, Washington

Date of application for amendment: December 17, 1998, as supplemented by letter dated January 21, 1999.

Brief description of amendment: The amendment revised Technical Specification Surveillance Requirement 3.8.1.8 to remove the restriction on testing of the manual transfer between the startup and backup offsite power sources while in Mode 1 or 2.

Date of issuance: January 27, 1999.

Effective date: January 27, 1999, to be implemented within 30 days from the date of issuance.

Amendment No.: 156.

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

Date of initial notice in ***Federal Register:*** December 22, 1998 (63 FR 70807).

The January 21, 1999, supplemental letter provided additional clarifying information and did not change the staff's original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 27, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Richland Public Library, 955 Northgate Street, Richland, Washington 99352.

Dated at Rockville, Maryland, this 3rd day of February 1999.

For the Nuclear Regulatory Commission.

John N. Hannon,

Acting Director, Division of Reactor Projects—III/IV, Office of Nuclear Reactor Regulation.

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NUCLEAR REGULATORY COMMISSION

Regulatory Guide; Issuance and Availability

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

Regulatory Guide 1.179, ''Standard Format and Content of License Termination Plans for Nuclear Power Reactors,'' has been developed to provide guidance on developing license termination plans for nuclear power reactor licensees who wish to terminate their licenses and release their sites.

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