from simplified operations and the reduced potential for undesirable opening of ERV will more than offset the reduction of the principal membrane safety factor. Reduced operational challenges will reduce the potential for undesirable impacts to the environment.

The change will not increase the probability or consequences of accidents, no changes are being made in the types of any effluents that may be released offsite, and there is no significant increase in the allowable individual or cumulative occupational radiation exposure. Accordingly, the Commission concludes that there are no significant radiological environmental impacts associated with the proposed action.

With regard to potential nonradiological impacts, the proposed action involves features located entirely within the restricted area as defined in 10 CFR Part 20. It does not affect nonradiological plant effluents and has no other environmental impact. Accordingly, the Commission concludes that there are no significant nonradiological environmental impacts associated with the proposed action.

Alternatives to the Proposed Action

Since the Commission has concluded there is no measurable environmental impact associated with the proposed action, any alternatives with equal or greater environmental impact need not be evaluated. As an alternative to the proposed action, the staff considered denial of the proposed action. Denial of the application would result in no change in current environmental impacts. The environmental impacts of the proposed action and the alternative action are similar.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the Final Environmental Statement for ANO–1.

Agencies and Persons Consulted

In accordance with its stated policy, on January 28, 1996, the staff consulted with the Arkansas State official, Mr. David Snellings, Director of the Division of Radiation Control and Emergency Management, regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

Based upon the environmental assessment, the Commission concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the Commission has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated November 26, 1996, which is available for public inspection at the Commission's Public Document Room, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Dated at Rockville, Maryland, this 7th day of March 1997.

For the Nuclear Regulatory Commission.

George Kalman,

Senior Project Manager, Project Directorate VI–1, Division of Reactor Projects III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 97–6342 Filed 3–11–97; 8:45 am]

BILLING CODE 7590–01–P

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

Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from February 14, 1997, through February 28, 1997. The last biweekly notice was published on February 26, 1997.

Notice of Consideration of Issuance of Amendments to Facility Opeating Licenses, Proposed No Significant Harzards Consideration determination, and Opportunity for a Hearing

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

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

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

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

By April 11, 1997, 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 aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1–(800) 248–5100 (in Missouri 1–(800) 342–6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to *(Project Director)*: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. 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.

Carolina Power & Light Company, et al., Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of amendment requests: December 4, 1996

Description of amendments request: The proposed amendments would revise the Technical Specifications (TS) to reflect a change in the method for detecting a reactivity anomaly described in TS 3.1.2 and TS Surveillance Requirement 4.1.2. Actual k_{eff} will be compared to predicted core k_{eff} instead of comparing actual and predicted control rod density to determine if a reactivity anomaly exists. Additionally, editorial changes to the Bases for TS 3/4.1.2 are proposed to support the TS amendments.

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 amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed license amendments modify the method of detecting a reactivity anomaly. The proposed license amendments allow using core k_{eff} to detect a reactivity anomaly instead of control rod density. The correlation between core

reactivity and control rod density depends on predicting core k_{eff}. Core k_{eff} can be readily monitored with the new plant process computer program and core k_{eff} can more accurately detect a reactivity anomaly in the core (assumptions are minimized). A reactivity anomaly is not considered an initiator of any previously analyzed accident. As such, changing the method of detecting a reactivity anomaly will not increase the probability of any accident previously evaluated. Although, a reactivity anomaly could impact the consequences of a previously analyzed accident, the consequences of an event occurring using the proposed method of detecting a reactivity anomaly are the same as the consequences of an event occurring using the current method of detecting a reactivity anomaly. As a result, the proposed amendments do not involve a significant increase in the consequences of any accident previously evaluated.

2. The proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed license amendments do not involve a physical modification to the plant. The proposed license amendments also continue to verify that the reactivity difference between predicted and actual are such that a reactivity anomaly does not exist. In addition, core k_{eff} can more accurately detect a reactivity anomaly in the core (assumptions are minimized) and can be readily monitored with the new plant process computer program. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed license amendments do not involve a significant reduction in a margin of safety. The proposed license amendments modify the method of detecting a reactivity anomaly. The proposed license amendments allow using core k_{eff} to detect a reactivity anomaly instead of control rod density. The correlation between core reactivity and control rod density depends on predicting core k_{eff}. Core k_{eff} can be readily monitored with the new plant process computer, and core k_{eff} can more accurately detect a reactivity anomaly in the core (assumptions are minimized). Therefore, the proposed license amendments 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: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403–3297.

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

NRC Project Director: Mark Reinhart, Acting

Carolina Power & Light Company, et al., Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant (BSEP), Units 1 and 2, Brunswick County, North Carolina

Date of amendment requests: January 7, 1997.

Description of amendments request: The proposed amendments would revise the Technical Specifications (TS) to: (1) exchange the reactor pressure vessel pressure-temperature (P–T) limits curves currently located in the Unit 1 and 2 TS; and (2) delete the current 8, 10, and 12 effective full power year (EFPY) hydrostatic test P–T limits curves and incorporate new 14 and 16 EFPY hydrostatic test P–T limits curves for the Unit 1 and 2 reactor pressure vessels. As reported in Licensee Event Report (LER) 1–94–05 dated March 22, 1994, and LER supplements dated April 29, 1994, and September 23, 1994, the licensee, the Carolina Power & Light Co. (CP&L), determined that the Unit 1 and 2 P–T limits curves had been inadvertently transposed and evaluated the effects of the transposition. The proposed amendments correct this transposition error. The proposed changes to the hydrostatic test P–T limits curves are required because it is anticipated that both units will exceed 12 EFPY during 1997.

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

This Technical Specification Change Request makes the following changes:

1. Exchanges the pressure-temperature limits curves currently located in the Unit 1 and Unit 2 Technical Specifications. In Licensee Event Report 1–94–05, CP&L reported that the Unit 1 and Unit 2 pressure-temperature limits curves had been inadvertently transposed. This request is an administrative change to relocate the pressure-temperature limits curves to Technical Specifications of the unit to which they correctly correspond.

2. Deletes the current 8, 10 and 12 effective full power year (EFPY) hydrostatic test pressure-temperature limits curves and incorporates new 14 and 16 effective full power year (EFPY) hydrostatic test pressure-temperature limits curves for the Brunswick Unit 1 and 2 reactors. The current reactor vessel pressure-temperature limits curves contained in the technical specifications for hydrostatic pressure tests are suitable for up to 12 effective full power years (EFPY) of reactor operation. It is anticipated that both units will surpass this threshold during 1997. Based on this, new pressure-temperature limits curves for 14 and 16 EFPY were developed. Commensurate changes to the references in Technical Specification 3/4.4.6.1 and Bases 3/4.6 are also proposed to reflect the deletion of current Technical Specification Figure 3.4.6.1–3c.

3. Reformat[s] the pressure-temperature limits curves in Technical Specification Figures 3.4.6.1–1, 3.4.6.1–2, 3.4.6.1–3a, and 3.4.6.1–3b. The changes associated with reformatting the Figures are administrative in nature.

Items 1, 2, and 3 do not involve a significant increase in the probability or consequences of an accident previously evaluated because of the following reasons:

1. Item 1 will exchange the Unit 1 and Unit 2 pressure-temperature limits curves. This change is considered administrative in nature. The pressure-temperature limits curves were developed based on design and materials information for the reactor vessel; however, due to an administrative error during the development of the curves, the materials information for the Unit 1 and Unit 2 reactor vessels was inadvertently reversed. Proposed change 1 is being made to exchange the reactor coolant system pressure-temperature limits curves. Therefore, since this proposed change does not involve a change to the pressure-temperature limits curves nor a change to the configuration of the facility, the probability of an accident previously evaluated is not increased.

Item 2 deletes the current Technical Specification hydrostatic test pressure-temperature limits curves and replaces them with updated curves. The current hydrostatic test pressure-temperature limits curves, which are valid through 12 EFPY are expected to expire during 1997; therefore, new hydrostatic test pressure-temperature limits curves were developed through 16 EFPY. These new hydrostatic test pressure-temperature limits curves will ensure that the integrity of the Brunswick Units 1 and 2 reactor pressure vessels is maintained during hydrostatic and leak tests up to 16 effective full power years of operation. The calculations used to generate the new pressure-temperature limits curves were performed using Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code, Welding Research Council Bulletin 175, and Appendix A to Section XI of the ASME Boiler and Pressure Vessel Code, and [incorporate] the requirements of 10 CFR 50, Appendix G, Section IV.A.2. For pressure-temperature limit curve development, the methods described in Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code are equivalent to the methods described in Appendix G to Section III of the ASME Boiler and Pressure Vessel Code. The proposed pressure-temperature limits curves, for hydrostatic and leak tests, take into consideration the effects of neutron irradiation on reactor vessel materials and provide the necessary margin, as specified by Appendix G of 10 CFR 50, to assure the structural integrity of the reactor coolant pressure boundary. Based on the above, it is concluded that this change will not increase the probability of an accident previously evaluated.

Item 3 reformats each of the Technical Specification Figures containing the pressure-temperature limits curves. The changes associated with the reformatting of proposed Technical Specification Figures 3.4.6.1–1, 3.4.6.1–2, 3.4.6.1–3a, and 3.4.6.1–3b reflect presentation preferences and do not result in technical changes (either actual or interpretational) to the requirements of the pressure-temperature limits curves. Therefore, the changes associated with reformatting the Technical Specification Figures containing the pressure-temperature limits curves are considered to be administrative in nature. Based on the above, it is concluded that this change will not increase the probability of an accident previously evaluated.

The proposed license amendments do not alter Limiting Safety System Settings nor Safety Limits. The proposed license amendments do not revise the technical bases from which the pressure-temperature limits curves were derived, and do not affect stresses and fatigue for transients and design basis events for which the reactor vessels were designed. The operation of plant equipment is not significantly impacted by the proposed license amendments. The proposed pressure-temperature limits curves provide the necessary margin to ... assure the structural integrity of the reactor coolant pressure boundary is maintained. This margin is designed to preclude the probability of a reactor coolant pressure boundary failure. In addition, since the proposed pressure-temperature limits curves are based on current regulatory requirements and fluence data, the consequences of a reactor coolant pressure boundary failure are not impacted by the proposed license amendments. Therefore, the proposed license amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed license amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed license amendments will ensure that acceptable pressure-temperature limits are imposed on the reactor pressure vessels during all phases of plant operation, thereby ensuring the structural integrity of the reactor pressure vessels. The pressure-temperature limits curves are designed to provide fracture protection for the reactor coolant pressure boundary and do not create any new accident modes. Accident modes for the reactor coolant pressure boundary, due to nonductile failure, are well understood by the industry. The proposed pressure-temperature limits curves and the Technical Specifications continue to provide controls to preclude such a failure. In addition, the proposed license amendments do not result in physical changes to the facility, nor do the proposed license amendments alter safety-related equipment, or safety functions. Therefore, the proposed license amendments do not create a new or different kind of accident from any previously evaluated.

3. The proposed license amendments do not involve a significant reduction in a margin of safety. The pressure-temperature limits curves are designed to provide a specific margin of safety. This margin is required to be at least as great as that specified in Appendix G to Section III of the ASME Boiler and Pressure Vessel Code and Appendix G to 10 CFR 50. The proposed pressure-temperature limits curves were developed based on design and materials information for the reactor vessels, current regulatory requirements and fluence data. The proposed pressure-temperature limit curves are based on analyses that ensure that the fracture toughness margins of 10 CFR Part 50, Appendix G are not exceeded. Therefore, the proposed license amendments do not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403–3297.

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

NRC Project Director: Mark Reinhart, Acting.

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

Date of amendment request: April 29, 1996, as supplemented on January 21, 1997.

Description of amendment request: The proposed amendment would:

1. Revise Technical Specification (TS) 3.7.1.1, Action a., to require the unit to be in hot shutdown, rather than cold shutdown, for consistency with NUREG–1431, ''Standard Technical Specifications for Westinghouse Plants,'' and add a new Action b. to clarify the shutdown requirements when there are more than three inoperable main steam line Code safety valves on any one steam generator.

2. Revise TS Surveillance Requirement 4.7.1.1 to clarify that Specification 4.0.4 does not apply for entry into Mode 3 for Byron and Braidwood and, for Braidwood only, delete the one-time requirements for Unit 1, Cycle 5 and Unit 2 after outage A2F27.

3. Revise the maximum allowable power range neutron flux high trip setpoints in Table 3.7–1.

4. Revise Table 3.7–2 to increase the as-found main steam safety valve (MSSV) lift setpoint tolerance to plus/minus 3%, provide an as-left setpoint tolerance of plus/minus 1%, and change a table notation.

5. Delete the orifice size column from Table 3.7–2.

6. Revise the Bases for TS 3.7.1.1 to be consistent with the proposed changes to TS 3.7.1.1.

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

The text describing reactor coolant loops and steam generators is redundant. TS 3.4.1.1, ''Reactor Coolant Loops and Coolant Circulation—Startup and Power Operation,'' and 3.4.1.2, ''Reactor Coolant Loops and Coolant Circulation—Hot Standby,'' provide restrictions on the number of operating reactor coolant loops and steam generators. Therefore, deleting the text that requires having four reactor coolant loops and associated steam generators in operation from TS 3.7.1.1, Action a., has no impact on any analyzed accident.

The proposed change to TS 3.7.1.1, Action a., to require the final mode to be hot shutdown rather than cold shutdown is consistent with the Applicability section of the specification, which does not require the MSSVs to be operable in hot shutdown. There are no credible transients requiring the MSSVs in modes 4 and 5. The steam generators are not normally used for heat removal in modes 5 and 6, and thus cannot be overpressurized. The change also eliminates the unnecessary transient that had been imposed on the unit by forcing entry into cold shutdown.

The new Action b. for TS 3.7.1.1 and text changes to Action a. clarify the shutdown requirement times based on the number of inoperable valves. There are no changes to these times.

Changing TSSR 4.7.1.1 to delete the one-time requirements imposed by previous amendments and allow entry into Mode 3 prior to performing the requirements of TSSR 4.0.5 has no impact on any accident. The change permits testing the MSSVs in accordance with the applicable codes and allows a reasonable amount of time for completion of the surveillance. The conditions requiring the one-time requirements have been corrected, so the one-time requirements are no longer required.

The proposed setpoints in Table 3.7–1 are more limiting than those currently allowed in Specification 3.7.1.1. Westinghouse determined that the current setpoints are non-conservative for some combinations of reduced MSSV availability and reactor power levels. By reducing the setpoints, the original design margins for safety will be met. Reduced reactor trip setpoints due to reduced availability of the MSSVs are not precursors to any accidents, but are used in the safety analysis to establish that plant response will be within required margins for accidents of concern.

Increasing the as-found valve setpoint tolerance from plus/minus 1% to plus/minus 3% does not have a significant impact on any accident. The peak primary and secondary pressures remain below 110% of design at all times. The departure from nucleate boiling ratio and peak cladding temperature values remain within the specified limits of the licensing basis. All of the applicable loss-of-coolant accident (LOCA) and non-LOCA design basis acceptance criteria remain valid.

The MSSVs are actuated after accident initiation to protect the secondary systems from overpressurization. Increasing the as-found setpoint tolerance will not result in any hardware modification to the MSSVs. Therefore, there is not an increase in the probability of the spurious opening of a MSSV. Sufficient margin exists between the normal steam system operating pressure and the valve setpoint with the increased tolerance to preclude an increase in the probability of actuating the valves. The MSSVs also remain capable of relieving any unlikely system overpressure during all applicable operating modes.

Although increasing the as-found valve setpoint tolerance may increase the steam release from the ruptured steam generator above the Updated Final Safety Analysis Review (UFSAR) value by approximately 2%, the steam generator tube rupture analysis indicates that the calculated break flow is still less than the value reported in the UFSAR. Therefore, the radiological analysis indicates that the slight increase in the steam release is offset by the decrease in the break flow such that the offsite radiation doses are less than those reported in the UFSAR. The evaluation also concluded that the existing mass releases used in the offsite dose calculation for the remaining transients (i.e., steam line break, rod ejection) are still applicable. Therefore, based on the above, there is no increase in the dose releases.

Neither the mass and energy release to the containment following a postulated LOCA, nor the analysis of containment response following the LOCA credit the MSSVs in mitigating the consequences of an accident. Therefore, changing the MSSV lift setpoint tolerances would have no impact on the containment integrity analysis. In addition, based on the conclusion of the transient analysis, the change to the MSSV tolerance will not affect the calculated steam line break mass and energy releases inside containment.

Deleting the orifice size column from Table 3.7.1–2 has no impact on previously evaluated accidents. There is no change to the orifice size, which is stated in the UFSAR and incorporated as needed in the accident analyses.

The proposed changes do not introduce any new equipment, equipment modifications, or any new or different modes of plant operation. The MSSVs are not precursors to any analyzed accident. The proposed changes will not affect the operational characteristics of any equipment or systems.

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

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

Deleting the text describing reactor coolant loops and steam generators from TS 3.7.1.1 Action a. has no impact on plant operation since the specific restrictions on the number of operating reactor coolant loops and steam generators are provided in TS 3.4.1.1 and 3.4.1.2.

The proposed change to TS 3.7.1.1, Action a., to require the final mode to be hot shutdown rather than cold shutdown is consistent with the Applicability section of the specification, which does not require the MSSVs to be operable in hot shutdown. There are no credible transients requiring the MSSVs in Modes 4 and 5. The steam generators are not normally used for heat removal in Modes 5 and 6, and thus cannot be overpressurized. NUREG–1431 does not include requirements for the MSSVs to be operable in these modes. The change will also eliminate the unnecessary transient that had been imposed on the unit by forcing entry into cold shutdown.

The new Action b. for TS 3.7.1.1 and text changes to Action a. clarify the shutdown requirement times based on the number of inoperable valves. There are no changes to the times.

The proposed change to TSSR 4.7.1.1 to clarify that TSSR 4.0.4 does not apply for entry into Mode 3 will allow ComEd to continue to perform MSSV testing at normal operating pressure and temperature as required by the applicable codes. The change precludes having to enter an action statement to perform the testing and eliminates severe time restrictions on the valve testing and conflicts with other plant startup requirements.

The proposed recalculated setpoints of Table 3.7–1 are more limiting than those currently allowed in the Specification and ensure that the original design margins for safety are met. The secondary system pressure remains within design limits.

Increasing the as-found tolerance on the MSSV setpoint to plus/minus 3% will not increase the challenge to the MSSVs or result in increased actuation of the valves. The changes to the Bases document the method for calculating the reduced reactor trip setpoints based on reduced availability of MSSVs.

Deleting the orifice size column from Table 3.7–2 and the obsolete one-time requirements in TSSR 4.7.1.1 are administrative changes only.

Increasing the lift setpoint tolerance on the MSSVs does not introduce a new accident initiator mechanism. The proposed change does not introduce any new equipment, equipment modifications, or any new or different modes of plant operation. No new failure modes have been defined for any system or component important to safety nor has any new limiting single failure been identified. This change will not affect the operational characteristics of any equipment or systems. Thus, there is no change in the margin for safety.

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

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

Deleting the text describing reactor coolant loops and steam generators has no impact on plant operation since the specific restrictions on the number of operating reactor coolant loops and steam generators are provided in TS 3.4.1.1 and 3.4.1.2.

The change requiring hot shutdown instead of cold shutdown entry is more appropriate than the existing specification since the action statement places the plant in a mode where operability of the MSSVs is not required. The Technical Specification is applicable in Modes 1, 2, and 3, therefore, entering Mode 4 places the plant in a condition where the MSSVs are not required to be operable. There are no credible transients requiring the MSSVs in Modes 4 and 5. The steam generators are not normally used for heat removal in Modes 5 and 6, and thus cannot be overpressurized. NUREG–1431 does not include requirements for the MSSVs to be operable in these modes.

Changing the mode in which the MSSVs are tested will not change the operational characteristics of the MSSVs. ComEd will continue to test the MSSVs at normal operating pressure and temperature as required by the applicable codes.

The proposed reactor trip setpoints in Table 3.7–1 are more limiting than the current setpoints in the Specification. Reactor trip settings were calculated using a revised methodology to account for the non-linear relationship of reactor trip setpoints and reduced MSSV availability. The revised setpoints ensure the original design margin of safety is maintained. The proposed changes to the Bases include the revised equation used to calculate the reduced reactor trip setpoints.

Increasing the as-found lift setpoint tolerance on the MSSVs will not adversely affect the operation of the reactor protection system, any of the protection setpoints, or any other device required for accident mitigation. The proposed increase in the setpoint tolerance does not invalidate the LOCA and non-LOCA conclusions presented in the UFSAR accident analyses. In letter CAE–91–209/CAE 91–219, Westinghouse concluded that the new loss of load/turbine trip analysis satisfied all applicable acceptance criteria and demonstrated that the conclusion presented in the UFSAR remains valid. For all the UFSAR non-LOCA transients, the departure from nucleate boiling design basis, primary and secondary pressure limits, and dose release limits continue to be met. Peak cladding temperatures remain well below the limits specified in the 10 CFR 50.46.

Deleting the orifice size column from Table 3.7–2 and the obsolete one-time requirements in TSSR 4.7.1.1 are administrative changes.

The proposed changes do not introduce any new equipment, equipment modifications, or any new or different modes of plant operation. These changes will not affect the operational characteristics of any equipment or systems. Therefore, no reduction in the margin of safety will occur as a result of 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 requested amendments involve no significant hazards consideration.

Local Public Document Room location: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Robert A. Capra.

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

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

Date of amendment request: August 23, 1996.

Description of amendment request: The proposed amendment would revise the technical specifications to reflect the design lineup for the Non-Accessible Area Exhaust Filter Plenum Ventilation System, and to make provisions for the performance of maintenance and testing on the 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:

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

Non-Accessible Area Exhaust Filter Plenum Ventilation (VA) System lineups are not considered as the precursors to any accident. The additional provisions added to the action statement for TS 3.7.7 accommodates required maintenance and surveillance activities. No new equipment is being installed and no existing equipment is being modified. Thus, these proposed changes will not result in an increase in the probability of occurrence of an accident previously evaluated.

On the postulated Loss Of Coolant Accident (LOCA) with Loss Of Offsite Power (LOOP), the operating plenum will either realign immediately or following the re-energization of its ESF bus which will occur within 10 seconds. Thus, there will always be at least one plenum operating immediately during an accident. The emergency procedures direct the realignment of the standby plenum. This direction is contained in the Byron and Braidwood Emergency Procedures (BEP/BwEP)–0, ''Reactor Trip or Safety Injection,'' and is performed prior to conducting event diagnostic steps.

Filtration of the air from the Emergency Core Cooling System (ECCS) equipment cubicles becomes critical when the ECCS pumps begin pumping accident water from the containment recirculation sumps. Prior to this the water flowing in these pumps is Refueling Water Storage Tank (RWST) water. This swap over from the RWST to the containment recirculation sump is expected to occur, at the earliest, 11 minutes following accident initiation leaving time to open the inlet damper on the standby VA plenum. Thus, since the standby plenum can be realigned before filtration of the ECCS equipment cubicle air is required, the Updated Final Safety Analysis Report (UFSAR) assumptions, and offsite dose calculation assumptions remain valid. There will be no significant change in the types or significant increase in the amounts of any effluent that may be released offsite, and there will be no significant increase in individual or cumulative occupational radiation exposure. Observations conducted on licensed operators undergoing simulator training verified that the VA system is realigned well before the swap-over to the containment recirculation sump under these conditions. Therefore, these proposed changes will not result in a significant increase in the consequences of an accident previously evaluated.

A review of the Byron and Braidwood Probabilistic Risk Assessment (PRA) shows that these proposed changes will have no effect on either Core Damage Frequency (CDF) or Uncontrolled Release Frequency (URF).

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

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

These proposed changes continue to ensure that, following a LOCA, the air being exhausted from the ECCS equipment rooms is properly filtered before being released to the environment.

These changes will not result in the installation of any new equipment or the modification of any existing equipment. No new operating modes or system interfaces will be created. The VA system will continue to operate as designed during normal and post accident conditions. All of the accident analysis assumptions and conditions will remain satisfied.

Thus this 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.

These proposed changes reflect the design lineup for the VA system and provide action requirements to accommodate required maintenance and surveillance testing. The VA system will continue to ensure that following a LOCA, the air being exhausted from the ECCS equipment rooms is properly filtered before being released to the environment.

Filtration of the ECCS equipment cubicle air does not become critical until the suction of the ECCS pumps is switched from the RWST to the containment recirculation sumps. This is postulated to occur, at the earliest, 11 minutes following accident initiation. On the postulated LOCA with LOOP, at lease one VA plenum will be in operation immediately and the emergency procedures direct the realignment of the standby plenum well before the ECCS pump suction swap-over. Observations conducted on licensed operators undergoing simulator training have verified this fact. Therefore, these proposed changes do not alter or affect any UFSAR or off-site dose calculation assumptions, and the margin of safety is not reduced.

A review of the Byron and Braidwood PRA shows that these proposed changes will have no effect on either CDF or URF.

No new equipment is being installed, and no existing equipment is being modified. The VA system will continue to operate as designed during normal and post accident conditions. All of the accident analysis assumptions remain satisfied.

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 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

Local Public Document Room location: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Robert A. Capra.

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

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

Date of amendment request: January 20, 1997.

Description of amendment request: The proposed amendment would change Technical Specification Table 3.6–1 to reflect planned changes in the plant configuration. As a result of the planned replacement of the Westinghouse D4 steam generators at Byron, Unit 1, and Braidwood, Unit 1, changes will be made to the containment isolation piping arrangements at the penetrations associated with the Feedwater (FW) and Auxiliary Feedwater (AF) systems. As a result of these changes, there will be no split FW flow with the replacement steam generators. AF flow will be fed into the main FW piping outside of containment and the existing FW tempering penetration will be used for a new steam generator recirculation system to be used during periods of extended shutdown. Additionally, since the replacement steam generators use a feedring design rather than a preheater design, the FW Isolation Bypass line and associated containment isolation valves will no longer be required. Table 3.6–1 of the Technical Specifications (TS) must be updated to reflect these changes. These changes do not affect the containment isolation capability originally designed to the criteria in 10 CFR 50, Appendix A, General Design Criteria (GDC) 54 through 57 as reflected in the Byron/Braidwood Updated Final Safety Analysis Report (UFSAR).

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.

Technical Specification 3/4.6.3 establishes the operability requirements for containment isolation valves as required by the Byron and Braidwood Operating Licenses in compliance with General Design Criteria 54 through 57 of Appendix A to 10 CFR 50. The operability of the containment isolation valves ensure that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere. Table 3.6–1 identifies these isolation valves and captures relevant information to ensure these valves remain operable under required conditions.

These proposed changes result in the elimination of the FW Isolation Bypass isolation valves. These isolation valves are not required with the replacement steam generator design. The remaining isolation valves have not been altered in any way, only the piping associated with them has been altered to the revised configuration. These changes do not result in alteration of any containment penetrations.

Failure of the piping between the isolation valve and the containment penetration is considered as an accident initiator. However, all piping changes between the isolation valve and the containment penetrations meet the requirements of the original design.

Therefore, since all original piping design criteria are met and the actual number of containment isolation valves is reduced, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

Each penetration identified in the proposed change is associated with a closed system inside containment and, as such, is provided containment isolation in accordance with the applicable requirements of GDC 54 through 57. There are four analyzed transients which take credit for feedwater isolation and are, therefore, relevant to this proposed change. These accidents are: (1) feedwater system malfunctions that result in an increase in FW flow, (2) inadvertent opening of a steam generator relief or safety valve, (3) steam system piping failure, and (4) FW system pipe break. All operability requirements for the affected containment isolation valves are unaffected by this proposed change.

The containment isolation valves' functions, system operating conditions, and accident responses are unchanged as a result of the new configuration. Therefore, since all original design criteria are met and each remaining isolation valve continues to provide the same degree of containment isolation as the original design, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

All modifications associated with the proposed changes will be outside of containment and can be characterized as the rearrangement of piping systems. All piping changes will comply with the original design of the plant and will retain required containment isolation capabilities per the requirements of GDC 54 through 57 as required by the current design basis. Piping configurations within the area of the containment penetration and the containment isolation valves are required to minimize branch connections per guidance in the Standard Review Plan (SRP) Section 3.6.2.

Therefore, since there are no unique configurations or reductions in design requirements, this proposed change does not create the possibility of any new or different kinds of accidents from those previously evaluated.

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

The proposed changes to the containment isolation arrangement are being made consistent with the same codes, standards, and isolation criteria as are currently in use at Byron and Braidwood. The containment isolation valves remaining in place following the steam generator replacement are unchanged with regard to their function, capability, reliability, or physical requirements. Containment isolation capability in accordance with GDC 54 through 57 is maintained at current levels of protection for the health and safety of the general public. Therefore, this proposed change does not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Robert A. Capra.

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

Date of amendment request: January 31, 1997.

Description of amendment request: The proposed amendments would revise the maximum allowable value in the Byron, Unit 1, Technical Specifications (TS), of the dose equivalent (DE) iodine-131 concentration in the primary coolant from the present value of 0.35 microcuries per gram of coolant to a maximum allowable of 0.20 microcuries per gram. This reduction in the DE iodine-131 concentration would be applicable only for the remainder of the present Byron, Unit 1, operating cycle (i.e., fuel cycle 8) which the licensee has previously stated will end in December 1997. The subject amendments are proposed by the licensee in order to provide additional margin with respect to the maximum Byron Station site allowable primary-to-secondary leakage limit from the Byron, Unit 1, steam generators (SG). This proposed Byron, Unit 1, TS revision to increase this margin is being proposed in conjunction with the proposed operating interval of 540 days above a T_{hot} temperature of 500 degrees Fahrenheit, between eddy current inspections (ECI) of the Byron 1 SGs. The last Byron, Unit 1, ECI was initiated in November 1995. This margin increase is being sought by the licensee to address staff concerns regarding potential SG tube leakage under postulated accident conditions due to SG tube circumferential cracking at the top of the tubesheet in the roll transition zone.

While the proposed revision to the DE iodine-131 is applicable only to Byron, Unit 1, the pending request for license amendments involves both Byron, Units 1 and 2, in that both units have a common set of TSs.

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

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

Generic Letter 95–05, ''Voltage Based Repair Criteria For Westinghouse Steam Generator Tubes Affected By Outside Diameter Stress Corrosion Cracking,'' allows lowering of the RCS DE I–131 activity as a means for accepting higher projected leak rates if justification for equivalent I–131 below 0.35 microcuries/gm is provided. Four methods for determining the impact of a release of activity to the public were reviewed to provide the justification.

They are as follows:

Method 1: NRC NUREG 0800, Standard Review Plan (SRP) Methodology
Method 2: Methodology described in a report by J.P. Adams and C.L. Atwood, ''The Iodine Spike Release Rate During a Steam Generator Tube Rupture,'' Nuclear Technology, Vol. 94 p. 361 (1991), using Byron Station reactor trip data.
Method 3: Methodology described in Adams and Atwood report, using normalized industry reactor trip data.
Method 4: Methodology described in draft EPRI Report TR–103680, Revision 1, November 1995, ''Empirical Study of Iodine Spiking in PWR Plants''.

The effect of reducing the RCS DE I–131 limit on the amount of activity released to the environment remains unchanged when the maximum site allowable primary-to-secondary leakage limit is proportionately increased. With a DE I–131 limit of 1.0 microcuries/gm, the maximum site allowable leakage limit was calculated in accordance with the NRC SRP methodology to be 12.8 gpm. The corresponding calculated activity released during a MSLB is 15.8 Ci. ComEd has evaluated the reduction of the DE I–131 to 0.20 microcuries/gm along with the increase of the allowable leakage to 64 gpm and has concluded:

—The maximum activity released is not changed, and
—The offsite dose including the iodine spiking factor is bounded by method 1.

Therefore, the offsite dose assessment and conclusions previously reached remain valid and continue to meet the requirements of 10 CFR 100.

An evaluation of Control Room dose attributed to a MSLB concurrent with steam generator primary-to-secondary leakage at the site allowable leakage limit was performed in support of a license amendment request for application of 1.0 volt Interim Plugging Criteria. This evaluation concluded that Control Room dose due to the MSLB scenario is bounded by the existing loss of coolant accident analysis. Therefore, the maximum site allowable primary-to-secondary leakage limit continues to be based on offsite dose at the Exclusion Area boundary due to MSLB leakage. This conclusion was previously submitted to the Staff in a September 22, 1994, transmittal in support of the 1.0 volt Interim Plugging Criteria license amendment request.

Based on the NRC SRP methodology for dose assessments, the Control Room dose, the Low Population Zone dose, and the dose at the Exclusion Area Boundary continue to satisfy the appropriate fraction of the 10CFR100 dose limits.

The Adams and Atwood report concluded that the NRC SRP methodology, which specifies a release rate spike factor of 500 for iodine activity from the fuel rod to the RCS, is conservative. In order to justify that a release rate spike factor of 500 is conservative, actual operating data from the previous reactor trips of Byron Unit 1 and Unit 2, with and without fuel failures, were reviewed and analyzed using the methodology presented Section II.C of the Adams and Atwood report (Method 2). The same five data screening criteria described in the Adams and Atwood report were applied to the Byron data to ensure consistency and validity when comparing the Byron results to the data in the Adams and Atwood report. Of the twenty-eight (28) reactor trip events at Byron Units 1 and 2, twelve (12) met the five data screening criteria.

Three of the Byron trips occurred during cycles with no failed fuel. In all three of these instances, the calculated spike factor was less than the spike factor of 500 assumed in the NRC SRP methodology. Byron, Unit 1, Cycle 8 is currently operating with no failed fuel and a DE I–131 activity of approximately 6E–4 microcuries/gm. The three previous trips with no fuel failures had steady-state iodine values that are relatively close to current operating conditions. It is therefore reasonable to conclude that the calculated spike factors from those trips would reflect the spike factor expected from an actual trip during the current cycle.

Based on the data in the Adams and Atwood report, the NRC SRP release rate spike factor of 500 may seem non-conservative since the Adams and Atwood factor was typically greater than 500 when initial concentrations were less than 0.3 microcuries/gm. The primary reason for these high ratios (up to 12,000) is not because the absolute post-trip release rate is high (factor numerator), but rather because the steady-state release rate (factor denominator) is low. The Byron specific data only resulted in one trip with a calculated release rate spike factor greater than 500, a value of 603.9. The trip occurred during the first operating cycle of Unit 2 which experienced failed fuel and a very low steady-state release rate. It is not expected based upon the current fuel cycle conditions that a spiking factor of greater than 500 would occur.

In order to compare the Byron specific data to the NRC SRP methodology, the release rate for a steady-state RCS DE I–131 activity of 1.0 microcuries/gm was calculated. Using the Byron specific data, the steady-state release rate is 17.6 Ci/hr. Using a release rate factor of 500 for the accident initiated spike, the post-trip maximum release rate would be 8797 Ci/hr. This is significantly higher than the largest iodine release rate of 127 Ci/hr from the Byron data. This demonstrates that, although a data point shows an iodine spike factor greater than 500, the resulting post-trip RCS DE I–131 fuel rod iodine release rate is less than the fuel rod iodine release rate from the NRC SRP methodology.

In the fourth method, the results from Draft EPRI Report TR–103680, Rev. 1, November 1995, ''Empirical Study of Iodine Spiking In PWR Power Plants'' were applied. The objective of the EPRI study was to quantify the iodine spiking in postulated Main Steam Line Break/Steam Generator Tube Rupture (MSLB/SGTR) sequences. In the EPRI report, an iodine spike factor between 40 and 150 was determined to match data from existing plant trips. The maximum iodine spike factor value of 150 was applied to a steady-state equilibrium RCS DE I–131 activity of 0.33 microcuries/gm. The resulting 2-hour average iodine concentration for a postulated MSLB/SGTR sequence was determined to be 3.1 microcuries/gm. Since the EPRI report is based on industry data and the EPRI method predicted a post-accident iodine activity which is a small fraction of the activity predicted by the NRC SRP methodology, it can be expected that, for the proposed 0.2 microcuries/gm limit under a MSLB/SGTR sequence, the post-accident iodine activity would be a small fraction of the RCS DE I–131 activity predicted by the NRC SRP methodology.

Lowering the Unit 1 RCS DE I–131 activity limit is conservative and remains bounded by the NRC SRP methodology. Thus, all offsite and control room dose assessment conclusions satisfy the appropriate limits of 10 CFR 100 and GDC 19. These proposed changes do not result in a significant increase in the consequences of an accident previously analyzed.

The RCS DE I–131 activity limit is not considered as a precursor to any accident. Therefore, this proposed change does not result in a significant increase in the probability of an accident previously analyzed.

The correction of the typographical error is administrative in nature and has no impact on either the probability or consequences of an accident previously analyzed.

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

The changes proposed in this amendment request conservatively reduce the Unit 1 DE I–131 limit at which action needs to be taken and correct a typographical error. The changes do not directly affect plant operation. These changes will not result in the installation of any new equipment or systems or the modification of any existing equipment or systems. No new operating procedures, conditions or modes will be created by this proposed amendment.

Thus, this 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.

NRC Generic Letter 95–05 allows lowering of the dose equivalent iodine as a means for accepting higher projected leakage rates provided justification for equivalent I–131 below 0.35 microcuries/gm is provided. Four methods for determining the fuel rod iodine release rates and spike factors during an accident were reviewed. Each of these methods utilized actual industry data, including Byron, Unit 1 and Unit 2, for pre- and post-reactor trip DE I–131 activities. Each of the methods demonstrated that the actual fuel rod iodine release rates are a small fraction of the release rate as calculated using the NRC SRP methodology. All design basis and off-site dose calculation assumptions remain satisfied. This proposed change will not result in a reduction in a margin of safety.

Correction of the typographical error is administrative in nature and does not impact the margin of safety. Therefore, the proposed changes do not result in a significant reduction in a margin of safety.

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

Local Public Document Room location: Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Robert A. Capra.

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

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

Date of amendment request: February 18, 1997.

Description of amendment request: The proposed amendment would revise Byron and Braidwood Technical Specification (TS) Table 2.2–1 (functional unit 13.a), ''Reactor Trip System Instrumentation Trip Setpoint: Steam Generator Water Level Low-Low'; TS Table 3.3–4 (functional unit 5.b.1), ''Engineered Safety Features Actuation System Instrumentation Trip Setpoints: Steam Generator Water Level-High-High'; TS Table 3.3–4 (6.c.1), ''Engineered Safety Features Actuation System Instrumentation Trip Setpoints: Steam Generator Water Level-Low-Low Start Motor-Driven Pump and Diesel-Driven Pump'; TS Surveillance Requirement (TSSR) 4.4.1.2.2, required steam generator inventory during hot standby; TSSR 4.4.1.3.2, required steam generator inventory during hot shutdown; and TS Section 3.4.1.4.1.b, limiting condition for operation during cold shutdown with loops filled.

The installation of Babcock and Wilcox International (BWI), replacement steam generators (RSGs) at Byron, Unit 1, and Braidwood, Unit 1, necessitates an increase to the operating range of the steam generators due to the decrease in narrow range span from 233 inches for the original Westinghouse Model D4 steam generators (OSGs) to 180 inches for the BWI RSGs. The increase in operating range will minimize the possibility of inadvertent plant trips following load changes and feedwater transients.

ComEd also proposes to eliminate notations from page 2–5 for both Braidwood and Byron and pages 3/4 3–25 and 3/4 3–26 (for Braidwood only) since they are related to cycles already completed and, therefore, are no longer valid.

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

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

This proposed change includes changing the low-low and high-high SG level setpoints. The setpoints are being changed to increase the SG level operating range. The change in acceptable operating range will decrease the possibility of inadvertent plant trips following load changes and feedwater transients. Therefore, the probability of inadvertent plant trips will decrease with this change.

The minimum setpoint change proposed in this request establishes controls to ensure that an adequate heat sink is maintained by providing an adequate secondary liquid mass to remove primary system sensible heat and core decay heat shortly after reactor trip and initiating auxiliary feedwater flow for long-term cooling. The accidents evaluated for this requirement are the Loss of Normal Feedwater and Feedwater Line Break transients.

The maximum setpoint ensures the steam lines and turbine remain undamaged from the introduction of low quality, two-phase flow from the steam generators into the steam lines. The accident evaluated for this requirement is the Feedwater System Malfunction that results in an increase in feedwater to one or more steam generators.

The steam generator water level setpoints are not considered a precursor to any of the analayzed accidents, and, therefore, these proposed changes do not result in an increase in the probability of occurrence of any accident previously analyzed.

The accidents evaluated for the low-low setpoint are the Loss of Normal Feedwater and Feedwater Line Break transients. These accidents were both analyzed using approved methodologies. All acceptance criteria were shown to be met for both these events. In addition, it was demonstrated that the Feedwater System Pipe Break response with the RSGs and the proposed low-low setpoint were bounded by the response with the original Model D4 steam generators. Therefore, the proposed low-low level setpoint change is demonstrated not to result in an increase in the consequences for these accidents.

The accident evaluated for the high-high setpoint is the Feedwater System Malfunction that results in an increase in feedwater to one or more Steam Generators. All acceptance criteria were shown to be met. In addition, it was shown that the RSGs do not completely fill with liquid. This assures that the steam lines and turbine remain undamaged with no introduction of low quality, two-phase flow from the steam generators into the steam lines during the transient. With all acceptance criteria met, the proposed high-high level setpoint change is demonstrated not to result in an increase in the consequences for these accidents.

TSSR 4.4.1.2.2, TSSR 4.4.1.3.2, and TS 3.4.1.4.1.b assure a minimum inventory (i.e., level) to provide decay heat removal. The requirement for a minimum inventory to remove decay heat is met with assurance that the tube bundle is completely covered. The steam generator operating water level during shutdown conditions are not considered a precursor to any accident, and, therefore, these proposed changes do not result in an increase in the probability of occurrence of any accident previously analyzed.

The elimination of outdated cycle specific notations from page 2–5 for both Braidwood and Byron and pages 3/4 3–25 and 3/4 3–26 (Braidwood only) are only administrative and does not impact the probability or consequences of any accidents previously analyzed.

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

The proposed setpoint changes do not create any new operating conditions or modes. The proposed change only revises the setpoints for the Reactor Trip System and Engineered Safety Features Actuation System. The actions of these systems will continue to be performed in accordance with existing requirements which are sufficient to ensure plant safety is maintained.

Shutdown conditions steam generator water level is necessary to assure adequate decay heat removal capacity. Assurance that the tube bundle is completely covered along with existing technical specification controls on the Auxiliary Feedwater System and on the Condensate Storage Tank ensure adequate heat removal capacity is maintained and that plant safety is maintained.

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

The elimination of outdated cycle specific notations from page 2–5 for both Braidwood and Byron and pages 3/4 3–25 and 3/4 3–26 (Braidwood only) are only administrative and does not create the possibility of a new or different accident.

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

A safety evaluation was performed to determine the effect of the RSGs with the revised setpoints.

The accidents potentially affected by the change in the Reactor Trip Steam Generator Water Level low-low setpoint (TS 2.2.1, Table 2.2–1, functional unit 13.a) and Engineered Safety Features Actuation System low-low AFW start setpoint (TS 3.3.2, Table 3.3–4, functional unit 6.c.1) are the Loss of Normal Feedwater and Feedwater Line Break transients. These accidents were both analyzed using approved methodologies. All acceptance criteria were shown to be met for both these events.

In addition, it was demonstrated that the Feedwater System Pipe Break response with the RSGs with the proposed low-low setpoint were bounded by the response with the OSGs. Therefore, the proposed low-low level setpoint change is demonstrated not to result in an reduction in the margin of safety for these accidents.

The accident potentially affected by the change in the Engineered Safety Features Actuation System high-high SG level trip (TS 3.3.2, Table 3.3–4, functional unit 5.b.1) is a Feedwater System Malfunction that results in an increase in feedwater to one or more steam generators. This accident was analyzed using an approved methodology. In the evaluation of the Feedwater System Malfunction, all acceptance criteria were shown to be met. In addition, it was shown that the RSGs do not completely fill with liquid. This assures that the steam lines and turbine remain undamaged with no introduction of low quality, two-phase flow from the steam generators into the steam lines during the transient. With all acceptance criteria met, the proposed high-high level setpoint change is demonstrated not to result in a reduction in the margin of safety.

There are no design basis accidents involving shutdown condition steam generator water level. Existing TS controls on the Auxiliary Feedwater System and on the Condensate Storage Tank ensure adequate heat removal capacity is maintained and that plant safety is maintained during shutdown conditions. Therefore, a change to the shutdown condition steam generator water level does not result in a reduction in the margin of safety.

The elimination of outdated cycle specific notations from page 2–5 for both Braidwood and Byron and pages 3/4 3–25 and 3/4 3–26 (for Braidwood only) are only administrative and does not result in a reduction in the margin of safety for any analyzed event.

Therefore, this amendment request does not result in a significant decrease in a margin of safety.

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

Local Public Document Room location: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603

NRC Project Director: Robert A. Capra.

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company,

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

Date of amendment request: January 31, 1997.

Description of amendment request: The proposed amendment will insert, by general reference, in the Perry Nuclear Power Plant Technical Specifications, the implementation document that the licensee will use to implement Option B, ''Performance-Based Requirements,'' to 10 CFR 50, Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.'' Option B to 10 CFR 50 Appendix J is an option that became effective on October 26, 1995.

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

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

The changes involved in this license amendment request revise the criteria for determining the Containment leak rate testing interval based upon past component performance. The revised criteria are based on the guidance contained in Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program.'' When the containment or containment penetrations have performed satisfactorily on a historical basis, this guidance permits the use of extended testing frequencies.

Since the allowable leakage rates are not being affected, the performance of the primary containment and systems and components penetrating the primary containment remains within acceptable limits. The functions and operation of these components will remain unchanged. Since the components are utilized to mitigate the consequences of accidents that require containment isolation, they are not considered to be accident initiators. Additionally, there are no accidents associated with implementation of a performance-based testing frequency for the primary containment and systems and components penetrating the primary containment.

As discussed previously, the components are utilized to mitigate the consequences of accident scenarios which rely upon the primary containment and systems and components penetrating the primary containment, to prevent the release of radioactive effluents. The implementation of Option B to 10 CFR 50 Appendix J is not intended to provide relief from the leakage criteria. The components will still be required to meet the leakage requirements as discussed in USAR Section 6.2.6 and Technical Specifications 3.6.1.1, 3.6.1.2, and 3.6.1.3. The primary containment isolation system is designed to limit leakage to L_a, which is defined by the Perry Technical Specifications to be 0.20 percent of primary containment air weight per day at the calculated peak containment pressure (P_a) for the design basis loss of coolant accident. The limitation on the rate of primary containment leakage is designed to ensure that the total leakage volume will not exceed the value assumed in the accident analyses at P_a. The L_a value is not being modified by this proposed change. Based on this, the primary containment and system and components penetrating the primary containment will remain capable of maintaining radioactive effluent releases within the limits of 10 CFR 100.

Because the proposed change does not alter the plant design, including the primary containment and primary containment penetrations, the proposed change does not directly result in an increase in primary containment leakage. Since the frequency will be based on the performance of the subject components, only those components that have satisfactorily maintained the actual leakage less than the allowable leakage will be tested less frequently. The testing frequency for components which have not satisfactorily limited leakage, or have not performed satisfactorily in the past, will not be altered. Other programs are also in place to ensure that proper maintenance and repairs are performed during the service life of the primary containment and systems and components penetrating the primary containment.

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

Several administrative/editorial changes have been incorporated (e.g., the clarification of the ''less than'' and ''less than or equal to'' signs on the Technical Specification acceptance criteria, and the retention of the standard frequency for the Drywell visual inspections). Such administrative/editorial changes do not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, these changes also do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a change to the plant design or operation, or new system interfaces. Consequently, the proposed change does not affect the parameters or conditions that could contribute to initiation of accidents. This change involves adopting a performance-based method for determining Type A, B, and C test frequencies. Except for the method of defining the test frequency, the methods for performing the actual tests are not changed. No new accident modes would be created by extending testing intervals. No safety related equipment or safety functions are altered as a result of this change. The change in testing frequency will not create any different types of accidents since the primary containment and systems and components penetrating the primary containment will continue to operate within their design bases. Therefore, reducing the test frequency would have no influence on, nor contribute to, the possibility of a new or different kind of accident or malfunction from those previously analyzed.

Based on the above discussions, the proposed change would not create the possibility of a new or different kind of accident than those previously evaluated.

The proposed administrative/editorial changes do 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. Thus, these changes also do not create the possibility of a new or different kind of accident from any previously evaluated.

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

This request does not involve a significant reduction in a margin of safety. The proposed change adopts a performance-based method for determining frequency of Type A, B, and C testing.

Except for the method of defining test frequency, no change in the method of testing is proposed. Since the frequency will be based on the performance of the subject components, only those components that have satisfactorily maintained actual leakage less than the allowable leakage will be tested less frequently. Other programs are also in place to ensure that proper maintenance and repairs are performed during the service life of the primary containment and systems and components penetrating the primary containment.

The margin of safety associated with the proposed change involves the offsite dose consequences of postulated accidents, which are directly related to the rate of primary containment leakage. The primary containment isolation system is designed to limit leakage to L_a, which is defined by the Perry Technical Specifications to be 0.20 percent of primary containment air weight per day at the calculated peak containment pressure (P_a) for the design basis loss of coolant accident. The limitation on the rate of primary containment leakage is designed to ensure that the total leakage volume will not exceed the value assumed in the accident analyses at P_a. The margin of safety for the offsite dose consequences of postulated accidents directly related to the primary containment leakage rate is maintained by continuing to meet L_a. The L_a value is not being modified by this proposed change. Based on this, the primary containment and systems and components penetrating the primary containment will remain capable of maintaining radioactive effluent releases within the limits of 10 CFR 100.

Therefore, the changes associated with this license amendment request do not involve a significant reduction in the margin of safety.

The proposed administrative/editorial changes will not reduce the margin of safety because they have no impact on safety analysis assumptions. These changes do not involve questions regarding safety issues, and therefore also 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: Perry Public Library, 3753 Main Street, Perry, Ohio 44081.

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

NRC Project Director: Gail H. Marcus.

Dairyland Power Cooperative (DPC), Docket No. 50–409, LaCrosse Boiling Water Reactor (LACBWR), Vernon County, Wisconsin

Date of amendment request: April 10, 1996.

Description of amendment request: This is a corrected notice that was first issued on August 1, 1996. The proposed amendment would update the facility Possession Only License and Technical Specifications to reflect the permanently shutdown and defueled condition of the plant. The amendment would also serve to remove the fire protection requirements, radiological effluent controls, quality assurance program controls and administrative controls for the emergency and security plans from the Technical Specifications to other inspectable and enforceable documents.

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:

DPC proposes to modify the LACBWR Technical Specifications to more accurately reflect the permanently shutdown, defueled, possession-only status of the facility.

Analysis of no significant hazards consideration:

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

The proposed changes delete system requirements that are no longer necessary to prevent, or mitigate the consequences of, a credible SAFSTOR accident as described in our current SAFSTOR Accident Analysis.

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

The proposed changes are either administrative in nature or were made based on the analysis of previously evaluated accident scenarios. In no other way do they change the design or operation of the facility and therefore do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The changes incorporate into the proposed Technical Specifications the margin of safety associated with the current SAFSTOR accident analysis and thus don't 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: LaCrosse Public Library, 800 Main Street, LaCrosse, Wisconsin 54601.

Attorney for licensee: Wheeler, Van Sickle and Anderson, Suite 801, 25 West Main Street, Madison, Wisconsin 53703–3398.

NRC Project Director: Seymour H. Weiss.

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

Date of amendment request: February 10, 1997 (TSC 95–04).

Description of amendment request: The proposed changes would revise the

Technical Specifications (TS) to reduce the allowable reactor building volume leakage rate per-day limit to permit removal of consideration of the penetration room contribution to the limit and the requirement to maintain the penetration room at a negative pressure with respect to all adjacent areas. Also, the penetration room ventilation system would be removed from the description of the containment in TS 5.2, and a surveillance requirement to perform a refueling outage test of the penetration room ventilation system would be added to TS 4.5.4. In addition, related changes would be made to the appropriate Bases sections.

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

No.

The following requirements are being removed from Technical Specifications regarding the PRVS [Penetration Room Ventilation System]:

(1) The requirement to measure reactor building leakage in excess of 50% of the total allowed containment leakage to the penetration room.

(2) The requirement, as specified in the design features, for the PRVS to maintain the penetration room at a negative pressure with respect to all adjacent areas. In addition, the design features description for the PRVS will be completely removed from Technical Specification 5.2 and replaced with a surveillance requirement in Technical Specification 4.5.4.

To demonstrate the inconsequential effects of the removal of the above requirements, a dose analysis was performed to conservatively demonstrate that PRVS adds margin, but is not necessary to meet 10CFR100 limits. The analysis assumes that the PRVS is completely unavailable for offsite dose reduction. However, the PRVS will be available, and all of the relevant operability and surveillance requirements for the PRVS will be retained in the Technical Specifications. Therefore, it is highly unlikely that the actual dose consequences would increase from 167 Rem thyroid to 240 Rem thyroid, since all surveillance and operability requirements for PRVS, other than the two requirements specified above, will be retained in Technical Specifications.

The specified Technical Specification requirements for PRVS are not accident initiators, nor will these requirements impact the probability of an accident. The purpose of these requirements is to ensure that the PRVS can reduce offsite dose to the public in the event of an accident which results in radioactive effluents leaking from the Reactor Building (RB) into the Penetration Room (PR).

In the initial ONS [Oconee Nuclear Station] design basis, the PRVS was credited to reduce offsite dose to the public in the event of certain accidents, such as a loss of coolant accident (LOCA) or Maximum Hypothetical Accident (MHA), where there is airborne leakage of radioactivity from the RB into the PR. The PRVS was credited to reduce the MHA two-hour Exclusion Area Boundary (EAB) dose to less than the 10CFR100 limit of 300 Rem thyroid. The current ONS dose analysis, which takes credit for the PRVS, calculates the MHA two-hour EAB dose to be 167 Rem thyroid. With a reduction in the allowable leakage from the Reactor Building (L_a) from 0.25 w%/day to 0.20 w%/day, while taking no credit for the PRVS, the two hour EAB MHA dose is calculated to be 240 Rem thyroid. This new dose analysis result meets the acceptance criterion of 10CFR100.

In addition to conducting a detailed dose analysis without taking credit for PRVS, a detailed review of PRA [probabilistic risk analysis] risk significance of the PRVS was conducted. The PRVS was determined to have virtually no PRA risk significance and no significant impact on consequences.

A review of the impact on control room habitability due to the proposed Technical Specification changes was conducted for credible UFSAR [Updated Final Safety Analysis Report] Chapter 15 accident scenarios. The operability requirements of the PRVS which are being retained in the Technical Specifications will ensure operability requirements are met to support the Control Room Ventilation System (CRVS). Therefore, removal of the identified statements pertaining to PRVS operability from Technical Specifications will not significantly impact control room habitability.

Based on the above information, the removal of the specified requirements for PRVS from Technical Specifications will not significantly increase the probability or consequences of an accident previously evaluated. The original design basis for offsite dose will still be met without any credit taken for the PRVS.

A change has been proposed to the Technical Specifications to reduce the allowable leakage from the Reactor Building (L_a) from 0.25 w%/day to 0.20 w%/day. This proposed change is conservative in nature since it will result in a potential reduction in the consequences of any accidents previously evaluated. Past integrated leak rate tests (ILRTs) for all three Oconee units have been reviewed by engineering and it has been concluded that this reduction in allowable leakage will have no impact on future station operation. This reduction is possible since the actual leakage of the ONS reactor buildings is far less than the original allowable design leakage.

B. Create the possibility of a new or different kind of accident from the accident previously evaluated?

No.

As stated previously, the proposed Technical Specification changes for the PRVS are not accident initiators, nor will these changes create the possibility of new or different kinds of accidents. The purpose of the PRVS is to reduce offsite dose to the public in the event of an accident which results in leakage from the RB into the PR.

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

C. Involve a significant reduction in a margin of safety?

No.

By reducing the allowable L_a to 0.20 w%/day, ONS meets 10CFR100 limits for off-site dose without taking any credit for the PRVS.

Although the margin to 10CFR100 limits is reduced by not taking credit for PRVS, it is concluded that the reduction in margin of safety is insignificant because:

(1) PRVS operability and surveillance requirements are being retained in Technical Specifications with the exception of two items which do not significantly degrade the ability of PRVS to perform its function.

(2) The reduction in the margin of safety is being offset by a reduction in L_a.

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: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691.

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

Date of amendment request: November 25, 1996.

Description of amendment request: The proposed amendment would allow decommissioning of the SNEF. The proposed changes to the license and technical specifications (TSs) would (1) accommodate decommissioning activities at the SNEF, (2) establish specific TS controls such as administrative controls and inspection requirements over decommissioning activities, (3) establish limiting conditions for performing decommissioning activities, (4) extend exclusion area controls to include the SNEF Decommissioning Support Building, (5) establish requirements for a Radiological Environmental Monitoring Program, an Off-Site Dose Calculation Manual and a Process Control Program, and (6) establish requirements for Technical and Independent Safety Reviews. In addition, the licensees have proposed other administrative and editorial changes to the TSs associated with the changes proposed above.

Basis for Proposed No Significant Hazards Consideration Determination: As required by 10 CFR 50.91(a), the licensees have provided their analysis of the issue of no significant hazards consideration, which is presented below:

The proposed changes do not involve a significant hazards consideration because the changes would not:

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

Accidents which might occur during the active decommissioning phase of the SNEF are bounded by the twelve accidents addressed in section 3.0 of the Updated Safety Analysis Report (USAR). The accident analyses addressed in the USAR demonstrate that no adverse public health and safety impacts are expected from accidents that might occur during decommissioning operations at the SNEF. The highest calculated dose to an individual located at the site boundary is less than 1.5 mrem to the whole body during a postulated materials handling accident. The dose to an individual located at the site boundary for other on-site accidents is at or below this value. The limiting accident case represents less than 0.15% of the EPA lower whole body dose limit for radiological accidents. Based on the analyses of postulated credible accidents that might occur during the planned decommissioning operations at the SNEF, it is concluded that no significant increase in the probability or consequences of an accident previously evaluated would be involved.

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

There are three general categories of accidents. These scenarios evaluate different methods of dispersing radioactive material to the environment which include a loss of support systems and external events. The first includes accident scenarios associated with decommissioning tasks. These were identified and evaluated as described in Section 3.0 of the USAR. The radiological effects of these accident scenarios are discussed in item 1 above. They do not, therefore, reflect a new or different kind of accident previously evaluated. The second category, loss of support systems, does not directly lead to an accident situation. Therefore, this category of event does not create the possibility of a new or different kind of accident. The final category of accidents involves external events.

Since these types of events can occur whether the SNEF is being decommissioned or not, the act of decommissioning does not create the possibility of a new or different kind of external event. Any potential radiological hazard that may occur as a result of an external event is addressed in item 1 above.

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

The TSs currently in place at the SNEF were developed to maintain a shutdown facility in a secured condition with occasional monitoring. These specifications were designed to ensure that the approximately 4 megacuries of radioactive material left on site following shutdown in 1972 as identified in the Saxton Decommissioning Plan and Safety Analysis Report dated April 1972, would remain safely contained. In the ensuing years, natural decay of these radioactive materials has resulted in a remainder of approximately 1500 curies of radioactive material at the facility (93% of which is activation contained within the steel structures of the reactor vessel). These proposed decommissioning TSs were developed in order to ensure this remaining radioactive material is safely contained and disposed of and that the environment surrounding the facility is monitored. These actions will assure that there is no reduction in the margin of safety during the active decommissioning of the facility. The final result of these efforts will be the removal of any potential radiological hazard from the site and the release of the site for unrestricted use.

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

Local Public Document Room Location: Saxton Community Library, Front Street, Saxton, Pennsylvania 16678.

Attorney for the Licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Project Director: Seymour H. Weiss.

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: January 28, 1997.

Description of amendment request: The proposed amendment would relocate the details of Technical Specification (TS) Section 6.2.3 on the Independent Safety Engineering Group (ISEG) from the Administration Controls section of the TSs and place these details in the Updated Final Safety Analysis Report (UFSAR) for South Texas Project, Units 1 and 2. This relocation is administrative only, and would not render any changes to the existing plant philosophy toward the ISEG or any safety analysis. Section 6.2.3 would be deleted from the TSs and removed from the table of contents for Administrative Controls. Currently UFSAR Section 13.4.2.2 describes the ISEG, but not in the detail as the current TSs.

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

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

The proposed changes move details from the Technical Specifications [TSs] to the Updated Final Safety Analysis Report (UFSAR). The changes do not result in any hardware or operating procedure changes. The details being removed from the Technical Specifications [TSs] are not assumed to be an initiator of any analyzed event. The UFSAR, which will contain the removed Technical Specification [TS] details, will be maintained using the provisions of 10 CFR 50.59 and is subject to the change control process in the Administrative Controls Section of the Technical Specifications [TSs]. [In addition] any changes to the UFSAR will be evaluated per 10 CFR 50.59, no increase in the probability or consequences of an accident previously evaluated will be allowed without prior NRC [Nuclear Regulatory Commission] approval. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes move details from the technical Specifications [TSs] to the Updated Final Safety Analysis Report (UFSAR). The changes will not alter the plant configuration (no new or different type of equipment will be installed) or make changes in methods governing plant operation. The changes will not impose different requirements, and adequate control of information will be maintained. The changes will not alter assumptions made in the safety analysis and licensing basis. Therefore, the changes 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.

The proposed changes move detail from the Technical Specifications [TSs] to the Updated Final Safety Analysis Report (UFSAR). The changes do not reduce the margin of safety since the relocation of details [is an administrative action and] has no impact on any safety analysis assumptions. In addition, the detail transposed from the Technical Specifications [TSs] to the UFSAR are the same as the existing Technical Specification [TS] [6.2.3]. [In addition] any future changes to the FSAR will be evaluated per the requirements of 10 CFR 50.59, no reduction in a margin of safety will be allowed without prior NRC approval. [Therefore, the licensee concluded that the changes will not involve a significant reduction in a margin of safety.]

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

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

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

NRC Project Director: William D. Beckner.

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

Date of amendment request: February 18, 1997.

Description of amendment request: The proposed amendment would change the reactor core fuel assembly design features requirements contained in Technical Specification 5.3.1, Fuel Assemblies. The proposed change would allow for the limited replacement of failed or damaged fuel rods in fuel assemblies with solid stainless steel or zirconium alloy filler rods in accordance with NRC-approved applications of fuel rod configurations. Reconstituted fuel assemblies would be limited to those fuel designs that have been analyzed with applicable NRC-staff-approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing would be allowed to be placed in nonlimiting core regions.

The proposed change would be in accordance with the guidance provided in NRC Generic Letter 90–02, Supplement 1, issued July 31, 1992.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below.

A. The changes do not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)) because the fuel assemblies would continue to meet the same fuel assembly and fuel rod design bases as the current fuel assemblies, the acceptance criteria for emergency core cooling systems would continue to be satisfied for all fuel assemblies, there would be no changes to reload design and safety analysis limits, and the radiological consequences of accidents previously evaluated would remain valid.

B. The changes do not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because the fuel assemblies would continue to satisfy the same design bases previously used. Since the original design criteria would be met, no new accident initiators would be introduced. All design and performance criteria would continue to be met for the use of reconstituted assemblies containing the approved filler rods. Furthermore, the use of reconstituted fuel assemblies does not affect the manner by which the facility is operated.

C. The changes do not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)) because the core reload design and safety analysis limits would be unchanged by the use of fuel assemblies containing approved filler rods. The use of all fuel assemblies would continue to be limited by the normal core operating conditions defined in the Technical Specifications. Reconstituted fuel assemblies would be evaluated specifically for each cycle reload core using approved reload design methods and approved fuel rod design models and methods.

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: Exeter Public Library, Founders Park, Exeter, NH 03833.

Attorney for licensee: Lillian M. Cuoco, Esquire, Northeast Utilities Service Company, Post Office Box 270, Hartford CT 06141–0270.

NRC Project Director: Patrick D. Milano.

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: March 4, 1996.

Description of amendment request: The proposed amendment would modify Surveillance Requirements 4.8.1.1.2.a.6, 4.8.1.1.2.b, and 4.8.1.1.2.g.7 by specifying load bands in loading the diesel generator (DG) in lieu of the present requirement to load the DG greater than or equal to a given value. A footnote is being added to the three surveillance requirements to indicate that a momentary transient outside the load range shall not invalidate the test. The associated Bases sections have been revised to reflect the above changes.

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

NNECO has reviewed the proposed changes in accordance with 10 CFR 50.92 and has concluded that the changes 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 purpose of the proposed changes to Surveillance Requirements 4.8.1.1.2.a.6, 4.8.1.1.2.b, and 4.8.1.1.2.g.7 is to provide the load bands for loading the DG during the monthly, 184 days and 18-month surveillances. Specifically, for monthly (Surveillance 4.8.1.1.2.a.6) and once per 184 days (Surveillance 4.8.1.1.2.b) surveillances, the load band is between 4800–5000 kW. For the 18-month surveillance (Surveillance 4.8.1.1.2.g.7), the load band is between 5400–5500 kW during the first 2 hours and between 4800–5000 kW during the remaining 22 hours. The specified load bands account for instrumentation inaccuracies using the plant computer and for the operational control capabilities and human factor characteristics. The proposed changes will keep the actual upper load limit of the DG below the manufacturer's recommended limit and the actual lower limit enveloping the accident load requirements. The proposed changes will reduce unnecessary engine stress and wear, while potentially improving overall diesel generator reliability and availability. The changes to the Bases section reflect the changes made to the surveillance requirements and, therefore, have no adverse impact on plant safety. Since the proposed changes serve to enhance overall safety, these changes do not increase the probability or consequences of any accident previously evaluated.

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

The proposed changes regarding the load band for the DGs do not affect the operation or response of any plant equipment, including the DG, or introduce any new failure mechanism. The proposed changes will reduce unnecessary engine stress and wear, while potentially improving overall DG reliability and availability. Therefore, the proposed changes do 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 changes specifying the load bands for diesel testing will keep the actual upper load limit of the DG below the manufacturer's recommended limit, and the actual lower limit enveloping the accident load requirements. Therefore, the proposed changes do not affect the capability of the diesel to perform its intended function. The purpose of these changes is to increase the overall DG reliability. The proposed changes do not impact the consequences of any design basis accidents. There is no direct impact on any of the protective boundaries. For these reasons, the changes do not involve a 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: 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, CT 06141–0270.

NRC Deputy Director: Phillip F. McKee.

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

Date of amendment request: January 31, 1997.

Description of amendment request: The amendments would revise Technical Specification 3/4.6.1.5, and its associated Bases section, to ensure that a representative average containment air temperature is measured.

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.

Limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a Loss of Coolant Accident or Steamline Break inside Containment. The resulting DBA temperature limits are used to established the environmental qualification envelope for safety-related electrical equipment inside containment.

The measurement of Containment average air temperature is a means to ensure that the design temperature normal operating limit is not exceeded. The probability of an accident is not impacted by the surveillance of normal temperature as it is a measurement which involves permanently installed, static equipment. The consequences of an accident are not impacted since the method of measurement ensures that the design basis temperatures are maintained and the intent of the existing surveillance specification is not changed. The proposed change does not impact the actual containment temperature, but specifies an acceptably accurate method for its determination.

Therefore, the probability of and consequences of an accident previously evaluated are not significantly increased.

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

The proposed changes do not involve any modifications to existing plant equipment, do not alter the function of any plant systems within Containment, do not introduce any new operating configurations or new modes of plant operation, nor change the safety analyses. The proposed change is consistent with NUREG–1431 and provides a methodology to ensure that calculated temperature is accurately determined.

The proposed changes will, therefore, 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 change results in an acceptably accurate determination of the containment average air temperature, therefore, compliance with the TS surveillance and its associated basis is assured. The present margin of safety is not affected since operating parameters and conditions are unchanged.

All changes are consistent with the intent of Salem's current TS and with the surveillance specified in NUREG–1431, Revision 1.

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: Mark J. Wetterhahn, Esquire, Winston and Strawn, 1400 L Street, NW, Washington, DC 20005–3502.

NRC Project Director: John F. Stolz.

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

Date of amendment request: February 11, 1997.

Description of amendment request: The amendments would add a new Technical Specification 3/4.7.10, ''Chilled Water System'' to address the support function this system provides to other necessary safety systems.

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

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

The Chilled Water System is a support system providing cooling to the Relay Rooms, the Control Room, and the affected Electrical Equipment Rooms. The Chilled Water System is not an accident initiator of any accident evaluated in the Safety Analysis Report. No physical changes to the Chilled Water System result from the proposed TS. The specified Allowed Outage Times in the TS are commensurate with the safety significance of the Chilled Water System as demonstrated by the PSA analysis.

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

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

The proposed changes do not involve any modifications to the Chilled Water System or mode of operation of the system. The proposed TS specifies the minimum operable number of chillers and chilled water pumps to assure that the system performs its design function. It does not change the basic way in which the Chilled Water System is operated. The loads that are isolated are non-safety loads. By maintaining the minimum operable number of chillers and chilled water pumps, adequate cooling is assured to the Relay Rooms, the Control Room, the affected Electrical Equipment Rooms.

Therefore, the 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.

The Chilled Water System is a support system which provides cooling to the Relay Rooms, the Control Room, and the affected Electrical Equipment Rooms. The proposed changes do not involve any modifications to the Chilled Water System or changes to the mode of operation of the system. The proposed TS establishes controls to better ensure that the Chilled Water System will be able to perform its intended design function and ensures that the safety functions of supported systems are maintained.

The proposed changes establish Allowed Outage Times and do not affect the operation of the Chilled Water System, and thus do not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Mark J. Wetterhahn, Esquire, Winston and Strawn, 1400 L Street, NW, Washington, DC 20005–3502.

NRC Project Director: John F. Stolz.

Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit No. 1, Ottawa County, Ohio

Date of amendment request: January 20, 1997.

Description of amendment request: The proposed amendment would change Technical Specification (TS) Section 3/4.5.2, ''Emergency Core Cooling Systems, ECCS Subsystems—$T_{avg} \geq 280$ °F,'' TS Section 3/4.5.3, ''Emergency Core Cooling Systems, ECCS Subsystems—$T_{avg} < 280$°F,'' and TS Section 3/4.7, ''Plant Systems.'' Several surveillance intervals would be changed from 18 months to once each refueling interval.

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:

Toledo Edison has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revisions to increase the surveillance test intervals from 18 to 24 months for the ECCS Subsystems (Surveillance Requirements 4.5.2.d.2.a, 4.5.2.e, 4.5.2.g.2, and 4.5.3), Auxiliary Feedwater System (Surveillance Requirement 4.7.1.2.1.c), Motor Driven Feedwater Pump System (Surveillance Requirement 4.7.1.7.d), Component Cooling Water System (Surveillance Requirement 4.7.3.1.b) and Service Water System (Surveillance Requirement 4.7.4.1.b). Initiating conditions and assumptions remain as previously analyzed for accidents in the DBNPS Updated Safety Analysis Report.

These revisions do not involve any physical changes to systems or components, nor do they alter the typical manner in which the systems or components are operated.

A review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because no potential for a significant increase in a failure rate of an affected system or component was identified during these reviews.

These proposed revisions are consistent with the NRC guidance on evaluating and proposing such revisions as provided in Generic Letter 91–04, ''Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle,'' dated April 2, 1991.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by these proposed revisions. Existing system and component redundancy is not being changed by these proposed changes. Existing system and component operation is not being changed by these proposed changes. The assumptions used in evaluating the radiological consequences in the DBNPS Updated Safety Analysis Report are not invalidated.

A review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because no potential for a significant increase in a failure rate of an affected system or component was identified during these reviews.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because these revisions do not involve any physical changes to systems or components, nor do they alter the typical manner in which the systems or components are operated. A review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because no potential for a significant increase in a failure rate of a system or component was identified during these reviews. No changes are being proposed to the type of testing currently being performed, only to the length of the surveillance test interval.

3. Not involve a significant reduction in a margin of safety because a review of the historical 18 month surveillance data and maintenance records identified no potential for a significant increase in a failure rate of a system or component due to increasing the surveillance test interval to 24 months. Existing system and component redundancy is not being changed by these proposed changes.

There are no new or significant changes to the initial conditions contributing to accident severity or consequences, therefore, there are no significant reductions 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: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, Ohio 43606.

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

NRC Project Director: Gail H. Marcus.

Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit No. 1, Ottawa County, Ohio

Date of amendment request: January 30, 1997.

Description of amendment request: The proposed amendment would change Technical Specification (TS) Section 2.2, ''Limiting Safety System Settings,'' and applicable bases, TS Section 3/4.3, ''Instrumentation,'' and applicable bases, TS Section 3/4.4, ''Reactor Coolant System,'' and TS Section 3/4.7, ''Plant Systems.'' Several surveillance intervals would be changed from 18 months to once each refueling interval. In addition, several setpoints would be revised based on an instrument drift study, and trip setpoints would be revised based on new calculations. Administrative revisions are also proposed consistent with these changes.

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

Toledo Edison has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revisions to increase the surveillance test intervals from 18 to 24 months for the subject Technical Specifications (TS): TS 2.2 Limiting Safety System Settings; TS 3/4.3.1.1, Reactor Protection System Instrumentation; TS 3/4.3.2.2, Steam and Feedwater Rupture Control System Instrumentation; TS 3/4.3.3.5.1, Remote Shutdown Instrumentation;

TS 3/4.3.3.6, Post-Accident Monitoring Instrumentation; TS 3/4.4.3, Safety Valves and Pilot Operated Relief Valve—Operating; TS 3/4.4.6.1, Reactor Coolant System Leakage Detection Systems; TS 3/4.7.1.2 and Auxiliary Feedwater System. Initiating conditions and assumptions remain as previously analyzed for accidents in the DBNPS Updated Safety Analysis Report.

Results of the instrument drift study analysis and review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because: the projected instrument errors caused by drift are bounded by the existing setpoint analysis or either a new analysis has been performed incorporating a more conservative setpoint or the calculations excess margin was reduced; projected instrument errors caused by drift are acceptable for control of plant parameters to effect a safe shutdown with the associated instrumentation or an engineering evaluation has been performed to justify continued use of the instrument string and revisions will be made to DBNPS calculations and controlling procedures where appropriate, to offset any adverse effect; and no potential for a significant increase in a failure rate of a system or component was identified during surveillance data and maintenance records reviews.

These proposed revisions are consistent with the NRC guidance on evaluating and proposing such revisions as provided in Generic Letter 91–04, ‘‘Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle,’’ dated April 2, 1991.

The proposed revisions to Allowable Values for Steam and Feedwater Rupture Control System Steam Generator Level—Low are conservative with respect to the current Allowable Values and therefore, do not adversely affect previously analyzed accidents.

The application of the Allowable Value to the Channel Functional Test only, the proposed deletion of the Trip Setpoint, and revision of the Limiting Condition for Operation and Action Statement A for TS 3.3.2.2, SFRCS Instrumentation, associated with the proposed revision of the Allowable Values for SFRCS Steam Generator Level—Low are consistent with NUREG–1430, Revision 1, ‘‘Standard Technical Specifications, Babcock and Wilcox Plants,’’ dated April, 1995. The proposed revisions will have no adverse effect on any previously analyzed accident.

The proposed revision to the Reactor Protection System High Flux Allowable Value was determined in accordance with the approved setpoint methodology described in Babcock and Wilcox document BAW–10179P, Safety Criteria for Acceptable Cycle Reload Analyses, and is bounded by the High Flux trip of 112% rated power assumed in the DBNPS accident analysis.

The proposed deletion of the Trip Setpoints, deletion of the Allowable Values applicable to the Channel Calibration for RC low pressure, and RC high pressure functional units, application of Allowable Values to the Channel Functional Test as opposed to the Channel Calibration, and deletion of the ‘‘**’’ and ‘‘#’’ footnotes for Technical Specification Table 2.2–1, Reactor Protection System Instrumentation Trip Setpoints, and the proposed revision to TS 2.2, Limiting Safety System Settings, are consistent with NUREG–1430, Revision 1, ‘‘Standard Technical Specifications, Babcock and Wilcox Plants,’’ dated April, 1995. The proposed revisions have no adverse effect on any previously analyzed accident.

The proposed revision to Technical Specification Table 4.3–10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 6, Containment Vessel Post-Accident Radiation separates the radiation monitors to reflect the revision to 24 month surveillance intervals for the High Range Radiation Monitors and that the Containment Wide Range Noble Gas monitors will remain on a 18 month surveillance frequency is an administrative change and does not affect previously analyzed accidents.

The proposed revision to the Technical Specification Bases 2.2.1, Reactor Protection System Instrumentation Setpoints, and Bases 3/4.3.1 and 3/4.3.2, Reactor Protection System and Safety System Instrumentation, are administrative and do not affect previously analyzed accidents.

Initiating conditions and assumptions remain as previously analyzed for accidents in the DBNPS Updated Safety Analysis Report.

These revisions do not involve any physical changes to systems or components, nor do they alter the typical manner in which the systems or components are operated.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by these proposed revisions. Existing system and component redundancy is not being changed by these proposed changes. Existing system and component operation is not being changed by these proposed changes and the assumptions used in evaluating the radiological consequences in the DBNPS Updated Safety Analysis Report are not invalidated.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because these proposed revisions do not involve any physical changes to systems or components, nor do they alter the typical manner in which the systems or components are operated.

No changes are being proposed to the type of testing currently being performed, only to the length of the surveillance test interval.

Results of the instrument drift study analysis and review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because: the projected instrument errors caused by drift are bounded by the existing setpoint analysis or either a new analysis has been performed incorporating a more conservative setpoint or the calculations excess margin was reduced; projected instrument errors caused by drift are acceptable for control of plant parameters to effect a safe shutdown with the associated instrumentation or an engineering evaluation has been performed to justify continued use of the instrument string and revisions will be made to DBNPS calculations and controlling procedures where appropriate, to offset any adverse effect; and no potential for a significant increase in a failure rate of a system or component was identified during surveillance data and maintenance records reviews.

The proposed revisions to Allowable Values for Steam and Feedwater Rupture Control System Steam Generator Level—Low are conservative with respect to the current Allowable Values and do not alter any testing currently being performed.

The application of the Allowable Value to the Channel Functional Test only, the proposed deletion of the Trip Setpoint, and revision of the Limiting Condition for Operation and Action Statement A for TS 3.3.2.2, SFRCS Instrumentation, associated with the proposed revision to the Allowable Values for SFRCS Steam Generator Level—Low are consistent with NUREG–1430, Revision 1, ‘‘Standard Technical Specifications, Babcock and Wilcox Plants,’’ dated April, 1995. The proposed revisions do not alter any testing currently being performed.

The proposed deletion of the Trip Setpoints, deletion of the Allowable Values applicable to the Channel Calibration for RC lowpressure, and RC high pressure functional units, application of Allowable Values to the Channel Functional Test as opposed to the Channel Calibration, and deletion of the ‘‘**’’ and ‘‘ι’’ footnotes for Technical Specification Table 2.2–1, Reactor Protection System Instrumentation Trip Setpoints, and the proposed revision to TS 2.2, Limiting Safety System Settings, are consistent with NUREG–1430, Revision 1, ‘‘Standard Technical Specifications, Babcock and Wilcox Plants,’’ dated April, 1995. The proposed revisions do not alter any testing currently being performed.

The proposed revision to the Reactor Protection System High Flux Allowable Value was determined in accordance with the approved setpoint methodology described in Babcock and Wilcox document BAW–10179P, Safety Criteria for Acceptable Cycle Reload Analyses, and is bounded by the High Flux trip of 112% rated power assumed in the DBNPS accident analysis and does not alter any testing currently being performed.

The proposed revision to Technical Specification Table 4.3–10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 6, Containment Vessel Post-Accident Radiation separates the radiation monitors to reflect the revision to 24 month surveillance intervals for the High Range Radiation Monitors and that the Containment Wide Range Noble Gas monitors will remain on a 18 month surveillance frequency is an administrative change and does not alter any testing currently being performed.

The proposed revision to the Technical Specification Bases 2.2.1, Reactor Protection System Instrumentation Setpoints, and Bases 3/4.3.1 and 3/4.3.2, Reactor Protection System and Safety System Instrumentation, are administrative and do not alter any testing currently being performed.

3. Not involve a significant reduction in a margin of safety because The results of the instrument drift study analysis and review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because: the projected instrument errors caused by drift are bounded by the existing setpoint analysis or either a new analysis has been performed incorporating a more conservative setpoint or the calculations excess margin was reduced; projected instrument errors caused by drift are acceptable for control of plant parameters to effect a safe shutdown with the associated instrumentation or an engineering evaluation has been performed to justify continued use of the instrument string and revisions will be made to DBNPS calculations and controlling procedures where appropriate, to offset any adverse effect; and no potential for a significant increase in a failure rate of a system or component was identified during surveillance data and maintenance records reviews. Existing system and component redundancy is not being changed by these proposed changes.

There are no new or significant changes to the initial conditions contributing to accident severity or consequences, consequently there are no significant reductions 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: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, Ohio 43606.

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

NRC Project Director: Gail H. Marcus.

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.

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

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

Date of amendment request: December 21, 1995, as supplemented on October 24, 1996.

Description of amendment request: The proposed amendments would relocate certain cycle-specific parameter limits from the Technical Specifications to the Operating Limits Report (ORL).

Date of publication of individual notice in Federal Register: February 20, 1997 (62 FR 7804).

Expiration date of individual notice: March 24, 1997.

Local Public Document Room location: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

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

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

Date of amendment request: November 5, 1996.

Description of amendment request: The proposed amendments would revise the technical specifications to allow ComEd to take credit, on a temporary basis, for soluble boron in the spent fuel storage water in maintaining an acceptable margin of subcriticality.

Date of publication of individual notice in Federal Register: February 10, 1997 (62 FR 6016).

Expiration date of individual notice: March 12, 1997.

Local Public Document Room location: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 1 and 2, Grundy County, Illinois

Date of amendment request: February 17, 1997.

Description of amendment request: The amendments would increase the maximum allowable water temperature for the Containment Cooling Service Water inlet and the Suppression Pool.

Date of publication of individual notice in Federal Register: February 27, 1997 (62 FR 8998).

Expiration date of individual notice: March 31, 1997.

Local Public Document Room location: Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450.

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.

Baltimore Gas and Electric Company, Docket No. 50–317, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Calvert County, Maryland

Date of application for amendment: October 3, 1996.

Brief description of amendment: The amendment concerns the provisions at Calvert Cliffs Unit 1 for receiving, possessing, and using byproduct, source, and special nuclear material. The amendment changed the Unit 1 license, which previously contained restrictions on the possession and use of byproduct, source, or special nuclear material, to be consistent with the Unit 2 license, which has no such restrictions. The staff found this license amendment to be acceptable since both units share the same radiation protection staff, and the training and procedures used to control the acceptance and use of radioactive material at Unit 2 are sufficient to control the radioactive material at Unit 1, as well.

Date of issuance: February 19, 1997.

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

Amendment No.: 220.

Facility Operating License No. DPR–53: Amendment revised the Operating License.

Date of initial notice in Federal Register: November 6, 1996 (61 FR 57482). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 19, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678.

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

Date of application for amendments: February 20, 1996 as supplemented October 16, 1996.

Brief description of amendments: The amendments revise Technical Specification (TS) 3.1.5, TS 3.1.10 and TS 4.1 to: (1) reduce the surveillance frequency for the boron concentration in the concentrated boric acid storage tank; (2) delete the surveillance requirements for Sr^{89} and $Sr^{90,}$ gross beta activity, gross alpha activity and dissolved gas concentration in the reactor coolant, and gross beta activity in the steam generator feedwater; (3) relocate the surveillance requirements for tritium, chloride, fluoride, and oxygen in the reactor coolant to the Selected Licensee Commitment (SLC) manual; and (4) delete TS 3.1.10 related to temperature and pressure requirements to avoid gas bubble formation on depressurization.

Date of issuance: February 19, 1997.

Effective date: As of the date of issuance to be implemented within 30 days. Implementation shall include concurrent revision of the Selected Licensee Commitment Manual in accordance with the application of this amendment.

Amendment Nos.: 221, 221, 218.

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

Date of initial notice in Federal Register: March 27, 1996 (61 FR 13523). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 19, 1997.

No significant hazards consideration comments received: No

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina.

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

Date of application for amendments: February 26, 1996.

Brief description of amendments: The amendments revise the TS to allow an increased limit for the nominal enrichment of new (unirradiated) Westinghouse-fabricated fuel stored in the new fuel storage racks.

Date of issuance: February 27, 1997.

Effective date: February 27, 1997, with full implementation within 45 days.

Amendment Nos.: 213 and 198.

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

Date of initial notice in Federal Register: April 24, 1996 (61 FR 18172) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 27, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085.

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

Date of amendment request: June 4, 1996 as supplemented by letter dated January 8, 1997.

Description of amendment request: The amendment revises Seabrook Appendix A Technical Specifications (TS) 1.7, ''Containment Integrity'', 3/4.6.1, ''Primary Containment'', and 3/4.6.5, ''Containment Enclosure Building'', to incorporate the provisions of Option B to 10 CFR Part 50, Appendix J. TS Section 6.15, ''Containment Leakage Rate Testing Program'', has been added to establish a Containment Leakage Rate Testing Program, as specified in Regulatory Guide 1.163, dated September 1995, to support these changes. In addition to the changes to incorporate the provisions of Option B, TS 3.6.1.7 and 4.6.1.7.1 have been revised to incorporate an increased leak testing interval and to include reference to the Containment Leakage Rate Testing Program.

Date of issuance: February 24, 1997.

Effective date: February 24, 1997.

Amendment No.: 49.

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

Date of initial notice in Federal Register: August 28, 1996 (61 FR 44359). The licensee's letter dated January 8, 1997, which provided additional information relating to containment purge supply and exhaust valve testing and maintenance, does not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 24, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 18, 1995.

Brief description of amendment: The amendment revises the Technical Specifications (TS) to extend the surveillance schedule from 18 months to each refueling interval (nominally 24 months) for TS 3/4.4.4, ''Relief Valves;'' TS 3/4.4.6.1, ''Reactor Coolant System

Leakage;'' TS 3/4.4.6.2, ''Operational Leakage;'' TS 3/4.4.9.3, ''Overpressure Protection Systems;'' and TS 3/4.4.11, ''Reactor Coolant System Vents.''

Date of issuance: February 19, 1997.

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

Amendment No.: 133.

Facility Operating License No. NPF–49: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: November 27, 1995 (60 FR 58402).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 19, 1997.

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 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut 06385

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

Date of application for amendments: October 25, 1996.

Brief description of amendments: The amendments revise the Technical Specifications (TSs) to incorporate the requirements of 10 CFR Part 50, Appendix J, Option B, for containment leakage tests. In addition, the amendments add a new section to the TSs, which establishes the requirements of the containment leakage rate testing program, consistent with the Improved Standard Technical Specifications.

Date of issuance: February 19, 1997.

Effective date: February 19, 1997, with full implementation within 30 days.

Amendment Nos.: 126 and 118.

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

Date of initial notice in Federal Register: January 15, 1997 (62 FR 2191) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 19, 1997.

No significant hazards consideration comments received: No.

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

Omaha Public Power District, Docket No. 50–285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: November 16, 1995, as supplemented by letter dated August 8, 1996.

Brief description of amendment: The amendment revises the technical specifications to add a limiting condition for operation and surveillance test for safety related inverters and deletes the nonsafety related instrument buses.

Date of issuance: February 13, 1997.

Effective date: February 13, 1997, to be implemented within 60 days from the date of issuance.

Amendment No.: 180.

Facility Operating License No. DPR–40. Amendment revised the Technical *Specifications.*

Date of initial notice in Federal Register: March 13, 1996 (61 FR 10395)

The August 8, 1996, supplemental letter provided additional clarifying information and 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 February 13, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102.

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: August 27, 1996.

Brief description of amendments: The proposed amendments change the minimum allowable charging water header pressure from a value of 955 psig to a value of 940 psig in Technical Specification 3.10.8, ''Shutdown Margin (SDM) Test-Refueling.''

Date of issuance: February 19, 1997.

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

Amendments Nos.: 218 and 221.

Facility Operating License Nos. DPR–44 and DPR–56: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: October 23, 1996 (61 FR 55036)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 19, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

Pennsylvania Power and Light Company, Docket Nos. 50–387 and 50–388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: February 2, 1996, as supplemented September 23, 1996.

Brief description of amendments: These amendments change Technical Specification 3.6.1.2 for each unit to permit primary containment leakage testing of the main steamline isolation valves at either 22.5 psig or 45 psig according to the type of test to be conducted.

Date of issuance: February 25, 1997.

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

Amendment Nos.: 163 and 134.

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

Date of initial notice in Federal Register: August 14, 1996 (61 FR 42282). The September 23, 1996, letter 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 February 25, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, PA 18701.

Southern California Edison Company, et al., Docket No. 50–362, San Onofre Nuclear Generating Station, Unit No. 3, San Diego County, California

Date of application for amendment: January 14, 1997.

Brief description of amendment: The amendment revises Surveillance Requirements (SRs) 3.8.1.14 and 3.8.1.15 to temporarily restore provisions of the emergency diesel generator surveillance requirements as they were prior to their revision as part of NRC Amendment No. 116 (conversion to the Improved Technical Specifications).

Date of issuance: February 10, 1997.

Effective date: February 10, 1997.

Amendment Nos.: 125.

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

Public comments requested as to proposed no significant hazards consideration: Yes (62 FR 3536 dated January 23, 1997). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by February 24, 1997, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 10, 1997.

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770.

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

Wisconsin Electric Power Company, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Unit Nos. 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of application for amendments: September 19, 1996, supplemented on November 18, 1996, revised on January 13, 1997, and supplemented on January 27, 1997.

Brief description of amendments: These amendments revise the reactor coolant system temperature below which the low temperature overpressure protection (LTOP) system and pressurizer power-operated relief valves (PORVs) shall be operable, modify the requirement to limit operation of the high pressure safety injection pump from reactor coolant system cold leg temperature of less than or equal to 275 °F to whenever the LTOP is required to be operable, change the name of the system from the overpressure mitigation system to the LTOP system, and revise the PORV setpoint from 425 psig to 440 psig.

Date of issuance: February 20, 1997, with full implementation within 45 days.

Effective date: February 20, 1997.

Amendment Nos.: 172 and 176.

Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes (62 FR 5256, dated February 4, 1997) The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided for an opportunity to request a hearing by March 6, 1997, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendments. The Commission's related evaluation of the amendments, finding of exigent circumstances, and final determination of no significant hazards considerations are contained in a Safety Evaluation dated February 20, 1997.

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

Local Public Document Room location: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

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

Date of amendment request: December 13, 1995, as supplemented by letter dated October 10, 1996.

Brief description of amendment: The amendment revises the 125-volt D.C. Sources (3.8.2.1 and 3.8.2.2) and Onsite Power Distribution (3.8.3.1 and 3.8.3.2) Technical Specifications to include provisions for installed spare battery chargers, which will be added to the plant design before startup from the ninth refueling outage.

Date of issuance: February 10, 1997.

Effective date: February 10, 1997, to be implemented before startup from the ninth refueling outage, currently scheduled to begin in September 1997.

Amendment No.: 104.

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

Date of initial notice in Federal Register: January 22, 1996 (61 FR 1639) The October 10, 1996, supplemental letter provided additional clarifying information and 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 February 10, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621

Notice of Issuance of Amendment to Facility Operating License and Final No Significant Hazards Consideration Determination

During the period since publication of the last biweekly notice, individual notices of issuance of amendments have been issued for the facilities as listed below. These notices were previously published as separate individual notices. They are repeated here because this biweekly notice lists all amendments that have been issued for which the Commission has made a final determination that an amendment involves no significant hazards consideration.

In this case, a prior Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing was issued, a hearing was requested, and the amendment was issued before any hearing because the Commission made a final determination that the amendment involves no significant hazards consideration.

Details are contained in the individual notice as cited.

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

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

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of

Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

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

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

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

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

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

For further details with respect to the action see (1) the application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By April 11, 1997, 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 aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1–(800) 248–5100 (in Missouri 1–(800) 342–6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (*Project Director*): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–001, 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 the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

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

Date of application for amendments: February 6, 1997, as supplemented February 12, 1997.

Brief description of amendments: The amendments revise Technical Specification 3.3.A to allow safety injection pump testing and evolutions during low-temperature shutdown conditions provided controls for reactor coolant system conditions are in place to provide low temperature overpressurization protection.

Date of issuance: February 20, 1997.

Effective date: February 20, 1997, with full implementation within 30 days.

Amendment Nos.: 127 and 119.

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

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. NRC published a public notice of the proposed amendments, issued a proposed finding of no significant hazards consideration, and requested that any comments on the proposed finding be provided to the staff by close of business on February 14, 1997. The notice was published in the Red Wing Republican Eagle on February 12, 1997, the Minneapolis Star Tribune on February 9, 1997, and the St. Paul Pioneer Press on February 10, 1997. No comments have been received.

The Commission's related evaluation of the amendments, finding of exigent circumstances, consultation with the State of Minnesota, and final determination of NSHC are contained in a Safety Evaluation dated February 20, 1997.

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

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

Dated at Rockville, Maryland, this 5th day of March 1997.

For The Nuclear Regulatory Commission.

Jack W. Roe,

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

[FR Doc. 97–5999 Filed 3–11–97; 8:45 am]

BILLING CODE 7500–01–P

OFFICE OF MANAGEMENT AND BUDGET

Interpretation Numbers 1 and 2 Related to Statement of Federal Financial Accounting Standards Numbers 4, 5, and 7

AGENCY: Office of Management and Budget.

ACTION: Notice of interpretations.

SUMMARY: This notice includes two interpretations of Statements of Federal Financial Accounting Standards (SFFAS), adopted by the Office of Management and Budget (OMB). These interpretations were recommended by the Federal Accounting Standards Advisory Board (FASAB) and adopted in their entirety by OMB.

FOR FURTHER INFORMATION CONTACT: Norwood J. Jackson, Jr. (telephone: 202–395–3993), Office of Federal Financial Management, Office of Management and Budget.

SUPPLEMENTARY INFORMATION: This Notice includes two interpretations of Statements of Federal Financial Accounting Standards (SFFAS), adopted by the Office of Management and Budget (OMB). These interpretations were recommended by the Federal Accounting Standards Advisory Board (FASAB) and adopted in their entirety by OMB.

Under a Memorandum of Understanding among the General Accounting Office, the Department of the Treasury, and OMB on Federal Government Accounting Standards, the Comptroller General, the Secretary of the Treasury, and the Director of OMB (the Principals) decide upon standards and concepts after considering the recommendations of FASAB. After agreement to specific standards and concepts, they are published in the Federal Register and distributed throughout the Federal Government.

An Interpretation is a document, originally developed by FASAB, of narrow scope which provides clarification of the meaning of a standard, concept or other related guidance. Once approved by the designated representatives of the Principals, they are published in the Federal Register.

This Notice, including the first two interpretations of SFFAS, is available on the OMB home page on the internet which is currently located at http://www.whitehouse.gov/WH/EOP/OMB/html/ombhome.html, under the caption ''Federal Register Submissions.''

G. Edward DeSeve,

Controller, Office of Federal Financial Management, Office of Management and Budget.

Interpretation Number 1 of Statement of Federal Financial Accounting Standards Number 7

Reporting on Indian Trust Funds in General Purpose Financial Reports of the Department of the Interior (DOI) and in the Consolidated Financial Statements of the United States Government: An Interpretation of SFFAS No. 7

Introduction

1. The DOI requested guidance about how to report information on Indian trust funds in the general purpose financial report of the Department. The Indian trust funds are managed by DOI's Office of Special Trustee, Office of the Secretary. (Prior to FY 1996, the trust funds were managed by the Bureau of Indian Affairs.) Some of the funds belong to individual Indians, others belong to tribes. The funds are managed by the Federal Government in a trust arrangement. While the government's responsibility for all of these funds is of a fiduciary nature, some portion of the annual flows for some of the funds have