Information Collection Request [paperwork package]. Comments submitted in response to this notice will be summarized and/or included in the request for Office of Management and Budget approval of the information collection request; they will also become a matter of public record.

Dated: December 9, 1997.

Adam M. Finkel,

Director, Directorate of Health Standards Programs.

[FR Doc. 97–32863 Filed 12–16–97; 8:45 am]

BILLING CODE 4510–26–M

NATIONAL ARCHIVES AND RECORDS ADMINISTRATION

Public Evaluation of NARA Archival Information Locator (NAIL)

AGENCY: National Archives and Records Administration (NARA).

ACTION: Notice.

SUMMARY: NARA is inviting the public to participate in an evaluation of its prototype online information system, the NARA Archival Information Locator (NAIL).

As part of its Electronic Access Project, NARA is constructing a nationwide, integrated online information delivery system. The project, a priority under the agency's Strategic Plan, will eventually result in a virtual card catalog of all NARA holdings nationwide, including those in the Presidential libraries and regional archives. In addition, copies of some of NARA's most popular and significant manuscripts, photographs, sound recordings, maps, drawings and other documents will be digitized and available for researchers to view online through the catalog.

To complete the final functional requirements for the catalog, NARA is undertaking an evaluation of its prototype, the NARA Archival Information Locator (NAIL). All members of the public are invited to use NAIL and to comment on its ease of use, functionality, and terminology.

NAIL can be accessed on the World Wide Web at http://www.nara.gov/nara/nail.html.

DATES: Comments should be received by January 31, 1998.

ADDRESSES: Comments can be sent through the online comments link in NAIL or by e-mail to nail.mailbox@arch2.nara.gov.

Dated: December 10, 1997.

L. Reynolds Cahoon,

Assistant Archivist for Human Resources and Information Services.

[FR Doc. 97–32914 Filed 12–16–97; 8:45 am]

BILLING CODE 7515–01–P

NUCLEAR REGULATORY COMMISSION

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 November 21, 1997, through December 5, 1997. The last biweekly notice was published on December 3, 1997 (62 FR 63970).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of 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 January 16, 1998, 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: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

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 28, 1997.

Description of amendment request: The proposed amendments would revise Byron and Braidwood Technical Specifications (TS) Sections 3/4.4.5, ''Steam Generators,'' and 3/4.4.8, ''Reactor Coolant System Specific Activity,'' for both the Byron Station, Units 1 and 2, and the Braidwood Station, Units 1 and 2. The intent of these proposed revisions is to restore for both Byron, Unit 1, and Braidwood, Unit 1, the original TS related to steam generator (SG) inspections and the primary coolant dose equivalent iodine-131 (DEI) concentrations. These amendments will become effective when the original steam generators (OSG) which are Westinghouse Model D4 SGs, are removed and the replacement steam generators (RSG) made by Babcock and Wilcox, International (BWI), are installed. The RSGs are presently being installed at Byron, Unit 1, while the RSGs will be installed at Braidwood, Unit 1, in fall 1998.

The SG inspection methodology, inspection frequency, reporting requirements and acceptance criteria for the RSGs in both Byron, Unit 1, and Braidwood, Unit 1, will revert to the TSs for the OSGs before several prior license amendments incorporated into the TSs: (1) The interim plugging criteria (IPC) consistent with Generic Letter (GL) 95–05; (2) the F* criteria for the SG tube expansions into the tubesheet; and (3) the criteria for repairing SG tubes using either Westinghouse laser welded sleeves or Combustion Engineering tungsten inert gas (TIG) welded sleeves. The TSs applicable to Byron, Unit 2, and Braidwood, Unit 2, both of which have Westinghouse Model D5 SGs, remain unchanged except for designating them in the TSs as model D5 SGs.

With respect to the limiting value of the DEI primary coolant concentration, both the Byron, Unit 1, TSs and the Braidwood, Unit 1, TSs will revert from their present TS limit of 0.35 to 1.0 microcuries per gram. A license amendment request to lower the Byron, Unit 1, TS DEI limit from 0.35 to 0.20 microcuries per gram was submitted on January 31, 1997, but this request was subsequently withdrawn on November 11, 1997, because the RSGs were being installed in the Byron, Unit 1, refueling outage which started in early November 1997. A license amendment request to lower the Braidwood, Unit 1, TS DEI limit from 0.35 to 0.10 microcuries per gram was submitted on September 2, 1997. Action on this request is still pending but in any case, will not affect the subject license amendment request for Braidwood, Unit 1, because the September 2, 1997, request is only applicable to the OSGs which are presently using the IPC that were originally incorporated into the TSs on November 9, 1995. The applicable bases sections of the Byron, Unit 1, TSs and Braidwood, Unit 1, TS will also be revised to reflect the TS changes discussed above.

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.

Due to design differences between the replacement Steam Generators (RSGs) and OSGs, the analyses supporting the application of the F* and voltage-based repair criteria do not apply to the RSGs. Also, the analyses supporting sleeving repair by the Westinghouse laser welded or Combustion Engineering Tungsten Inert Gas (TIG) welded sleeving methodologies do not apply to the RSGs due to the design differences. The RSG and OSG tube bundle configurations are similar, however, the RSG tubes are smaller in diameter, constructed of Inconel Alloy 690 instead of Alloy 600, and supported by stainless steel lattice grids instead of the drilled carbon steel plates used in the OSGs. The RSG tubes are hydraulically expanded into the tube sheet during initial assembly. The RSG upper tube bundle shape consists of tubes with continuous, smooth, long radius bends.

The structural analysis demonstrates that the tube integrity is maintained for a Main Steamline Break (MSLB) occurring during normal full power operation. The structural evaluation of the tubing for faulted conditions was performed in accordance with the ASME Boiler and Pressure Vessel Code Section III requirements. The tube material selection and size exceed the strength requirements of the existing steam generators. Comparison of the Alloy 690 tube material used in the RSGs with the Alloy 600 tube material in the OSGs show that the RSG material strength characteristics are as good as or better than those of the existing design. A comparison of the stress margins of the RSG and OSG show that the stress margin in the RSG tubes exceed the stress margin in the OSG tubes.

RSG portions of the reactor coolant pressure boundary are designed to permit periodic inspection and testing of important areas and features to assess structural and leak-tight integrity. ASME Section XI, provides the depth of an allowable outside diameter (O.D.) flaw for tubes in service. The RSG has tubing fabricated from SB–163 material (Inconel Alloy 690) which is examined by eddy current methods to the requirements of ASME Section III, NB–2550. The tubing has a radius to thickness (r/t) ratio less than 8.70. In accordance with ASME Section XI, for tubing having an r/t ratio of less than 8.70, the depth of an allowable O.D. flaw shall not exceed 40% of the nominal tube wall thickness.

The potential for tube rupture is not increased from the OSGs as demonstrated in the qualification analysis and testing for the RSGs. The program for periodic inservice inspection of the steam generators monitors the integrity of the SG tubing to ensure that there is sufficient time to take proper and timely corrective action if any tube degradation is detected. Therefore, installation of the RSGs will not increase the probability of the occurrence of primary-to-secondary leakage or a steam generator tube rupture (SGTR) during normal or accident conditions.

The design basis doses calculated for postulated accidents involving degradation of SG tubes, such as SGTR and MSLB accidents, as presented in UFSAR [Updated Final Safety Analysis Report] Chapter 15 accident analysis have been evaluated and are decreased by installation of the RSGs and restoration of the RCS activity limit to 1.0 microcuries/gm. The decrease in offsite dose is primarily due to the smaller RSG tube diameter and less primary-to-secondary transfer during the event. The dose calculations are performed consistent with NUREG–0800, ‘‘Standard Review Plan’’ and ensure site boundary doses are within a small fraction of the Title 10 Code of Federal Regulations Part 100 (10 CFR 100) requirements. Therefore, the change does not involve a significant increase in the consequences of an accident previously evaluated.

Limiting the applicability of TS provisions to a specific cycle or SG type are administrative changes in that they provide clarification consistent with current analyses and do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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

Restricting application of IPC, F* and sleeving methodologies to the OSGs and reinstating an RCS activity limit of 1.0 microcuries/gm upon installation of the RSGs will not introduce significant or adverse changes to the plant design basis that could lead to a new or different kind of accident being created. The RSG tubing meets the requirements of General Design Criteria (GDC) 14, 15, 30, 31, and 32 of 10 CFR 50, Appendix A. The RSG tubing has been designed and evaluated consistent with ASME Code Section III criteria and the inspection criteria for the RSGs is consistent with ASME Code Section XI criteria. The RSGs have thermally treated Inconel Alloy 690 tubes which are hydraulically expanded into the tube sheet during initial assembly. Alloy 690 is more resistant to stress corrosion cracking (SCC) than Alloy 600 which is used in the OSG tubing. Overall tube bundle structural and leakage integrity is maintained at a level consistent with or better than the originally supplied tubing during all plant conditions.

ComEd will continue to apply the TS maximum primary-to-secondary leakage limit of 150 gpd (0.1 gpm) through any one SG at Byron and Braidwood to help preclude the potential for excessive leakage during all plant conditions. The EPRI recommended 150 gpd limit provides for leakage detection and plant shutdown in the event of an unexpected tube leak and precludes the potential for excessive leakage or tube burst in the event of a Main Steam Line Break or under Loss of Coolant Accident conditions.

Limiting the applicability of TS provisions to a specific cycle or SG type are administrative changes in that they provide clarification consistent with current analyses.

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

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

Restricting application of IPC, F*, and sleeving methodologies to the OSGs for which the supporting analyses apply, does not involve a reduction in a margin of safety. The RSG tubing has been shown to retain adequate structural and leakage integrity during normal, transient, and postulated accident conditions consistent with GDC 14, 15, 30, 31, and 32 of 10 CFR 50 Appendix A. The RSG tubing has been designed and evaluated consistent with the margins of safety specified in ASME Code Section III. The proposed program for periodic inservice inspection of the replacement steam generators monitors the integrity of the SG tubing to ensure that there is sufficient time to take proper and timely corrective action if any tube degradation is present. The proposed program is consistent with the Standard Technical Specifications.

The Unit 1 RCS dose equivalent I–131 limit is being raised upon installation of the RSGs to eliminate the compensatory lower limit that was adopted in conjunction with IPC for the existing Westinghouse D4 SGs. With the RCS activity limit returned to the Standard Technical Specification value of 1.0 [mu]Ci/gm, the assessment of postulated UFSAR Chapter 15 accidents (including SGTR and MSLB) has concluded that the calculated design basis doses presented in Chapter 15 are not adversely impacted by the RSGs. This ensures that the resulting 2-hour dose rates at the Byron and Braidwood site boundaries will not exceed an appropriately small fraction of 10 CFR 100 dose guideline values.

Limiting the applicability of TS provisions to a specific cycle or SG type are administrative changes in that they provide clarification consistent with current analyses.

Therefore, it is concluded that this change does not involve a significant reduction in a margin of safety with respect to plant safety as defined in the UFSAR or the Technical Specification.

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.

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

Date of amendment request: March 26, 1997.

Description of amendment request: The proposed amendment would revise the containment system technical specifications (TS) contained in TS Sections 3.6 and 4.5. The licensee has classified the changes as ''More Restrictive,'' ''Less Restrictive,'' and ''Administrative.'' ''More Restrictive'' changes include reduction of the allowable containment pressure, addition of an action statement defining action to be taken when the containment pressure limit is exceeded, addition of a restriction on containment temperature, and revision of the applicable conditions for the containment purge valves to require that the valves be operable above 210 degrees F versus the current requirement that they be operable above 525 degrees F. ''Less Restrictive'' changes include addition of an allowance to enter an air lock through a locked door to perform maintenance, addition of an allowance to open containment isolation valves under administrative control, revision of the applicable conditions for containment pressure to exclude the cold shutdown operating condition, and addition of an exception to the surveillance requirement requiring verification of the status of ''locked-closed'' manual isolation valves after a refueling outage to exclude requiring such verification for valves opened under administrative control. ''Administrative'' changes include the deletion of containment isolation valve tables and component identifiers from the TS in accordance with Generic Letter 91–08 (''Removal of Component Lists from Technical Specifications'') and editorial restructuring of the affected TS sections to clarify the remaining requirements.

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

Each proposed change has been classified as ''Administrative,'' ''More Restrictive,'' or ''Less Restrictive.'' ''Administrative'' and ''More Restrictive'' changes are discussed generically; ''Less Restrictive'' changes are discussed individually.

Five of the proposed changes are classified as being ''Less Restrictive'':

(G.1) Allowance in LCO [Limiting Condition for Operation] 3.6.1 to enter an air lock to perform maintenance.

(G.2) Allowance in LCO 3.6.1 to open containment isolation valves under administrative control.

(I.2) Revising the applicable conditions of LCO 3.6.2, Containment Pressure to exclude Cold Shutdown.

(J.2) Exception in SR [Surveillance Requirement] 4.5.3d for valves opened under administrative control as allowed by LCO 3.6.1.

(P) Allowance in SR 4.5.2 to enter an air lock to perform maintenance.

Four of the proposed changes are classified as being ''More Restrictive'':

(I.1) Revising LCO 3.6.2 to reduce the allowable containment pressure.

(I.3) Addition of an action statement to LCO 3.6.2, Containment Pressure.

(K) Addition of a new LCO which restricts Containment Temperature.

(M.2) Revising the applicable conditions for LCO 3.6.5, Purge Valves.

The remaining changes are all classified as being ''Administrative''.

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

1. Changes G.1, G.2, J.2, and P: Proposed changes G.1 and P allow limited access through the operable door of an air lock when the other door is inoperable; current Technical Specifications [TS] do not. Proposed changes G.2 and J.2 allow unisolating containment penetration flow paths intermittently under administrative control; current TS do provide a similar allowance, but only for one specific penetration. These changes cannot significantly increase the probability of an accident because opening an air lock door or a containment penetration is not, itself, an initiator and does not affect the items which are initiators of any analyzed accident.

The ability to open the operable door or to open a containment penetration, even if it means the containment boundary is temporarily not intact, does not significantly increase the consequences of an accident previously evaluated because of the low probability of an event that could pressurize the containment occurring during the short time the operable door or containment penetration is expected to be open. In a case where containment integrity (or containment operability) is lost due to excessive leakage, both the Palisades Technical Specifications and the Standard Technical Specifications [STS] allow one hour of continued operation for its restoration. That time period is allowed without regard to the magnitude of the potential leakage, and would be allowed even if both personnel air lock doors [were] leaking excessively. The additional allowance of permitting the operable door to be opened momentarily for entry or egress when the other door is inoperable due to excessive leakage would not significantly add to the probability of containment leakage and the resultant consequences of an accident. Similarly, the allowance to open any containment penetration intermittently under administrative control, which currently is allowed for one penetration, would not significantly add to the probability of containment leakage and the resultant consequences of an accident.

Therefore, operation of the Facility in accordance with proposed changes G.1, G.2, J.2, and P would not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Change I.2: Change I.2 alters existing LCO 3.6.2, Containment Pressure so that it no longer applies during Cold Shutdown. LCO 3.6.2 is intended to limit containment pressure to that value used as an initial condition in the safety analysis. Containment pressure is an initial condition in analyses which assure that containment internal pressure will not exceed the containment design values during a LOCA or MSLB. Containment pressure is not an initiator of any accident previously evaluated. Neither a LOCA [loss-of-coolant accident] nor a MSLB [main steam line break] occurring during Cold Shutdown would pressurize the containment. Therefore, a containment pressure LCO is not necessary, during Cold Shutdown, to assure that containment design pressure and temperature is not exceeded. The STS Containment pressure LCO is not applicable in Cold Shutdown.

Therefore, operation of the Facility in accordance with proposed change I.2 would not involve a significant increase in the probability or consequences of an accident previously evaluated.

3. More Restrictive Changes: ''More Restrictive'' changes only add new requirements, or revise existing requirements to result in additional operational restrictions. The TS, with all ''More Restrictive'' changes incorporated, will still contain all of the requirements which existed prior to the changes. Therefore, ''More Restrictive'' changes cannot involve a significant increase in the probability or consequences of an accident previously evaluated.

4. ''Administrative'' changes make wording changes which clarify existing TS requirements, without affecting their

technical content. Since ''Administrative'' changes do not alter the technical content of any requirements, they cannot involve a significant increase in the probability or consequences of an accident previously evaluated.

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

1. Changes G.1, G.2, J.2, and P: Proposed changes G.1 and P allow limited access through the operable door of an air lock when the other door is inoperable; current Technical Specifications do not. Proposed changes G.2 and J.2 allow unisolating containment penetration flow paths intermittently under administrative control; current TS do provide a similar allowance, but only for one specific penetration. Opening an air lock door or a containment penetration does not affect the operating conditions or operation of any plant systems (other than the containment); it does not create a threat to the integrity of any operating system or alter any system operating practice or settings.

Since the opening of an air lock door or a containment penetration only affects the potential leakage from the containment, and does not affect any of the operating plant systems, operation of the Facility in accordance with the proposed Technical Specifications change would not create the possibility of a new or different kind of accident from any previously evaluated.

2. Change I.2: Change I.2 alters existing LCO 3.6.2, Containment Pressure so that it no longer applies during Cold Shutdown. LCO 3.6.2 is intended to limit containment pressure to that value used as an initial condition in the safety analysis. Containment pressure is an initial condition in analyses which assure that containment internal pressure will not exceed the containment design values during a LOCA or MSLB. Neither a LOCA nor a MSLB occurring during Cold Shutdown would pressurize the containment. Therefore, a containment pressure LCO is not necessary, during Cold Shutdown, to avoid creation of a new or different kind of accident. The STS Containment pressure LCO is not applicable in Cold Shutdown.

Therefore, operation of the Facility in accordance with proposed change I.2 would not create the possibility of a new or different kind of accident from any previously evaluated.

3. More Restrictive Changes: ''More Restrictive'' changes only add new requirements, or revise existing requirements to result in additional operational restrictions. The TS, with all ''More Restrictive'' changes incorporated, will still contain all of the requirements which existed prior to the changes. Therefore, ''More Restrictive'' changes cannot create the possibility of a new or different kind of accident from any previously evaluated.

4. ''Administrative'' changes make wording changes which clarify existing TS requirements, without affecting their technical content. Since ''Administrative'' changes do not alter the technical content of any requirements, they cannot create the possibility of a new or different kind of accident from any previously evaluated.

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

1. Changes G.1, G.2, J.2, and P: Proposed changes G.1 and P allow limited access through the operable door of an air lock when the other door is inoperable; current Technical Specifications do not. Proposed changes G.2 and J.2 allow unisolating containment penetration flow paths intermittently under administrative control; current TS do provide a similar allowance, but only for one specific penetration. The ability to open the operable door or a containment penetration, even if it means the containment boundary is temporarily not intact, does not involve a significant reduction in a margin of safety because of the low probability of an event that could pressurize the containment occurring during the short time the operable door or penetration is expected to be open.

Therefore, operation of the Facility in accordance with the proposed Technical Specifications change would not involve a significant reduction in a margin of safety.

2. Change I.2: Change I.2 alters existing LCO 3.6.2, Containment Pressure so that it no longer applies during Cold Shutdown. LCO 3.6.2 is intended to limit containment pressure to that value used as an initial condition in the safety analysis. Containment pressure is an initial condition in analyses which assure that containment internal pressure will not exceed the containment design values during a LOCA or MSLB. Neither a LOCA nor a MSLB occurring during Cold Shutdown would pressurize the containment. Therefore, elimination of a Cold Shutdown LCO for containment pressure would not affect the post-accident pressure or temperature. Since peak post accident [pressure] and temperature would be unaffected by the proposed change, operation of the Facility in accordance with proposed change I.2 would not involve a significant reduction in a margin of safety.

3. More Restrictive Changes: ''More Restrictive'' changes only add new requirements, or revise existing requirements to result in additional operational restrictions. The TS, with all ''More Restrictive'' changes incorporated, will still contain all of the requirements which existed prior to the changes. Therefore, ''More Restrictive'' changes cannot involve a significant reduction in a margin of safety.

4. ''Administrative'' changes make wording changes which clarify existing TS requirements, without affecting their technical content. Since ''Administrative'' changes do not alter the technical content of any requirements, they cannot 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: Van Wylen Library, Hope College, Holland, Michigan 49423.

Attorney for licensee: Judd L. Bacon, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Project Director: John N. Hannon.

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 amendments request: October 29, 1997.

Description of amendments request: The proposed amendments to the Technical Specifications (TS) for the Brunswick Steam Electric Plant (BSEP) Units 1 and 2 would revise the description of the control rod assemblies (CRAs) in TS 5.3.2. The proposed revision was requested to support replacement of a portion of the BSEP Unit 1 CRAs during that unit's next refueling outage with assemblies of a different design. Carolina Power & Light Company, the licensee, has proposed adopting the description of CRAs used in NUREG–1433, Revision 1, ''Standard Technical Specifications General Electric Plants, BWR/4,'' which includes the number and shape of CRAs and a stipulation that NRC-approved absorber material be used in CRAs. The more detailed description in the current TS of CRAs would be relocated to the Updated Final Safety Analysis Report. The licensee has stated that the CRA description proposed for TS 5.3.2 will be sufficient to ensure that any future changes in CRA design that may affect safety will require prior NRC review and approval.

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.

Relocation of the control rod assembly descriptive information from the Technical Specifications to the Updated Final Safety Analysis Report will ensure that adequate control of the information is maintained. Any changes to this design information must conform with the requirements of 10 CFR 50.59. Restricting use of control rod assembly absorber materials to those listed, or to materials that have been approved by the NRC, will ensure any changes which may affect safety to require prior NRC review and approval. Since the information with a potential to affect safety is sufficiently addressed by the Technical Specifications, the criteria of 10 CFR 50.36(c)(4) for including the relocated information as Design Features are not met. Because the relocated information is not required to be in the Technical Specifications to provide adequate protection of the public health and safety, relocation of control rod assembly descriptive information will not increase either the probability or the consequences of an 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.

Relocation, to the Updated Final Safety Analysis Report, of the information pertaining to the control rod assembly designs ensures that adequate control of the information will be maintained. Since the information with a potential to affect safety is sufficiently addressed by the Technical Specifications, the criteria of 10 CFR 50.36(c)(4) for including the relocated information as Design Features are not met. Because the relocated information is not required to be in the Technical Specifications to provide adequate protection of the public health and safety, the proposed Technical Specification changes to relocate the control rod assembly design information to the Updated Final Safety Analysis Report 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.

As discussed in Items 1 and 2 above, relocation of the control rod assembly descriptive information from the Technical Specifications to the Updated Final Safety Analysis Report will ensure that adequate control of the information is maintained. Any changes to this design information must conform with the requirements of 10 CFR 50.59. Restricting use of control rod assembly absorber materials to those listed, or to materials that have been approved by the NRC, will ensure any changes which may affect safety to require prior NRC review and approval. The information with a potential to affect safety is sufficiently addressed by the Technical Specifications, therefore, the proposed Technical Specification changes to relocate control rod assembly design information to the Updated Final Safety Analysis Report 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: James E. Lyons.

GPU Nuclear Corporation, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: July 16, 1997, as supplemented October 30, 1997.

Description of amendment request: The amendment would update License condition 2.C(4) to reflect the latest revision levels of the Oyster Creek Security Training and Qualification Plan, License Amendment Request No. 252.

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:

GPU Nuclear has concluded that the proposed changes to the Security Plan do not involve a significant hazard consideration. In support of this determination, an evaluation of each of the three standards set forth in 10 CFR 50.92 is provided below.

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

Security Plan provisions are not associated with design basis accident initiators nor do they constitute part of any mitigation system. Therefore, the probability and consequences of accidents are not increased.

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

The Security Plan changes do not create new or change existing physical interfaces with plant equipment. Therefore, the changes do not create the possibility of a new or different kind of accident.

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

Margins associated with reactor and fuel storage nuclear safety are not affected by the proposed Security Plan changes since neither physical nor procedural changes to associated systems, structures and components are involved. Vital area security measures, which are reduced, are compensated by commitments to hold contingency drills at a frequency sufficient to maintain response capability for response personnel and to use organic-type X-ray equipment.

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

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

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

NRC Project Director: Ronald B. Eaton, Acting Director.

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: November 14, 1997.

Description of amendment request: The proposed change to Technical Specification 4.5.2.d.1 will clarify the wording and increase the setpoint for the open pressure interlock (OPI).

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 revision in accordance with 10 CFR 50.92 and has concluded that the revision does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not satisfied. The proposed revision does not involve [an] SHC because the revision would not:

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

Increasing the Technical Specification Open Pressure Interlock (OPI) pressure to 412.5 psia [pounds per square inch—atmospheric] will still maintain the required function of preventing the MOVs [motor operated valves] from opening inadvertently. The increased pressure is within the design limits of the RHR [residual heat removal] piping system and components. The pressure signal is generated from a transmitter and results in an electronic input to the bistable. This is a clarification of the conditions under which the OPI is tested.

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

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

There is no change to the function of the OPI. The protection provided by the interlock remains intact. The Technical Specification OPI pressure has been raised to take into account instrument accuracies and reset deadbands. The RHR system design pressure remains protected from being exceeded by inadvertent opening of the isolation MOVs. The method for the OPI surveillance is clarified by clearly stating that the bistable receives a simulated transmitter signal representative of the process pressure.

Therefore, the proposed revision does 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 design pressure of the RHR system is 600 psig [pounds per square inch—gauge]. The most limiting case is to prevent the RHR pump developed head pressure from exceeding the design pressure when aligned to the RCS [reactor coolant system] as suction pressure. RHR pump testing has determined that a maximum pump differential pressure of 195 psi [pounds per square inch] exists for deadhead/no flow conditions. Therefore, to maintain the 600 psig design pressure limit, RCS/suction pressure must be limited to 405 psig (420 psia, assuming a 15 psi conversion from psig to psia). The proposed maximum pressure, including setpoint tolerances and reset deadbands, is less than this value; i.e. 412.5 psia. Head corrections due to elevation differences are considered to be insignificant.

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

In conclusion, based on the information provided, it is determined that the proposed revision does not involve an SHC.

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

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

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

NRC Deputy Director: Phillip F. McKee.

Power Authority of The State of New York, Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: November 3, 1997.

Description of amendment request: The proposed amendment would revise Sections 1, 3.1, 3.3, 4.3, and 6 of Appendix A of the Indian Point 3 Technical Specifications. These revisions extend the Heatup-Cooldown limits from 11 to 13 effective full power years (EFPYs), provide the corresponding Overpressure Protection System (OPS) limits, relocate the new pressure temperature limit curves and low-temperature overpressurization protection (LTOP) system limits to the pressure temperature limit report (PTLR) and include some minor revisions which ensure specification clarity and conservatism.

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

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

Response: The proposed license amendment does not involve a significant increase in the probability or consequences of a previously analyzed accident. The pressure-temperature limit changes proposed by this amendment are based on supporting data and evaluation methodologies previously submitted to the NRC in Reference 3 [see application dated November 3, 1997] and approved as Amendments 109 and 121 (References 4 and 5) [see application dated November 3, 1997]. These limits are based upon the irradiation damage prediction methods of Regulatory Guide 1.99, Revision 2. The LTOPs changes contained in this submittal have been conservatively adjusted in accordance with the new pressure-temperature limits, in accordance with the methodology contained in Reference 3 and ASME Code Case N–514.

The relocation of the pressure-temperature and LTOPs limits from the Technical Specifications to the PTLR does not eliminate the requirement to operate in accordance with the limits specified in 10 CFR [Part] 50, Appendix G. The requirement to operate within the limits in the PTLR is specified in and controlled by the Technical Specifications.

The revised version of Section 3.1.A.8 clarifies existing requirements related to the OPS system and adds an eight hour completion time for compensating actions, consistent with the STS [standard technical specifications]. The changes to Section 3.1.A.1.h, i, and j revise the requirements associated with the start of an RCP [reactor coolant pump]. These changes improve specification clarity and do not increase the probability or consequences of an accident.

The Technical Specification changes associated with the restriction on SI [safety injection] pumps provides added conservatism to the Technical Specifications and limits the likelihood of an RHR [residual heat removal] overpressurization event. Current plant procedures prohibit actuation of any SI pumps when RHR is in service, except during testing, loss of RHR cooling, or reduced inventory operations. Therefore, the change to the Technical Specifications will not alter current plant operation.

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

Response: The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously analyzed. The pressure-temperature limits are updating the existing limits by taking into account the effects of radiation embrittlement, utilizing criteria defined in Regulatory Guide 1.99, Revision 2, and extending the effective period to 13 EFPYs. The updated OPS limits have been adjusted to account for the effect of irradiation on the limiting reactor vessel material. These changes do not affect the way the pressure-temperature or OPS limits provide plant protection and no physical plant alterations are necessary. The relocation of the pressure-temperature and OPS limits from the Technical Specifications to the PTLR does not alter the requirements associated with these limits.

The revisions to Section 3.1.A.8 concerning the OPS system improve on the clarity of existing specifications and add a completion time for compensating actions that is consistent with the STS. These changes do not involve any hardware modifications and do not affect the function of the OPS system.

The revisions concerning the operation of SI pumps bring the Technical Specifications into line with current operating procedures. The changes to Specification 3.1.A.1.h, i, and j provide specification clarity and are more conservative than existing Technical Specifications. Therefore, the changes cannot create the possibility of a new or different kind of accident.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response: The proposed amendment does not involve a significant reduction in a margin of safety. The margins of safety against fracture provided by the pressure-temperature limits are those limits specified in 10 CFR Part 50, Appendix G and ASME Boiler and Pressure Vessel Code Section XI, Appendix G. The guidance in these documents has been utilized to develop the pressure-temperature limits with the requisite margins of safety for the heatup and cooldown conditions. The new LTOP limits are based upon Reference 3 and ASME Code Case N–514. The relocation of the pressure-temperature and OPS limits to the PTLR does not alter the requirements associated with these limits.

The revisions to Section 3.1.A.8 clarify the requirements associated with the OPS system. The revisions associated with the operation of SI pumps with RHR in service (Sections 3.3.A.8, 9 and 10) and the changes regarding RCP starts (Section 3.1.A.1.h, i, and j) are more conservative than the current Technical Specifications, and are consistent with plant operating procedures. Therefore, they do not reduce a margin of safety.

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

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

Attorney for licensee: Mr. David Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Project Director: S. Singh Bajwa.

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: October 24, 1997.

Description of amendment request: The amendments would increase the containment hydrogen analyzer surveillance frequency in Technical Specification 4.6.4.1 from once per refueling outage to quarterly.

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 containment hydrogen analyzers provide control room indication of hydrogen concentration in the containment atmosphere. They do not affect the probability of any previously evaluated accident. The proposed change would increase the calibration frequency specified in TS 4.6.4.1 to make it consistent with manufacturer's recommendations and the current calibration frequency at [Salem Generating Station] SGS as imposed by administrative controls. The change in TS-required calibration frequency is in the conservative (more frequent) direction, to ensure that potential degradation of the sensor electrolyte over time would not result in unacceptable performance of the hydrogen analyzers. The change in specified frequency would not adversely affect the consequences of any previously evaluated accident.

2. Proposed change does not create the possibility of a new or different kind of accident from any accident previously analyzed [evaluated].

The proposed change affects only the specified calibration frequency of the containment hydrogen analyzers. The proposed change does not affect the design of any SGS structure, system or component, nor would it result in any new plant configuration. Therefore, it does not create the possibility of a new or different kind of accident.

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

The proposed change to the containment hydrogen analyzer calibration frequency does not affect the design or operating limits of any SGS structure, system or component. The change would make the specified calibration frequency more conservative, to ensure the hydrogen analyzers perform as designed over time. The proposed change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: 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: October 24, 1997.

Description of amendment request: The amendments would revise Technical Specification (TS) 3/4.7.7, ''Auxiliary Building Exhaust Air Filtration System.'' The revisions would: (1) Require both Auxiliary Building Ventilation (ABVS) supply fans to be operable, (2) require all three ABVS exhaust fans to be operable, (3) align ABVS TSs to be consistent with current TS bases and recently revised system descriptions in the Salem Updated Final Safety Analysis Report (UFSAR), (4) assure that negative pressure is maintained in the Auxiliary Building under all postulated single active failures, (5) clarify required Engineered Safety Feature filter testing, (6) provide consistency between Unit 1 and Unit 2 TSs, and (7) for Unit 2 only, remove the requirement to verify safety injection auto-start capabilities.

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 change alters the number of fans which must be OPERABLE to ensure that a sufficient number of supply and exhaust fans will be operable, following a most limiting single failure, to mitigate the consequences of design basis accidents. The changes to the ABVS surveillance requirements still provide an appropriate means for demonstrating the operability of the ABVS.

The ABVS cannot initiate or otherwise cause any accident or operational transient evaluated in the UFSAR. Consequently, the probability of such events is not increased. The ABVS cannot increase the consequences of a design basis LOCA unless: (1) Auxiliary Building negative pressure is lost, resulting in uncontrolled, ground level release of radioactive material; (2) ABVS carbon adsorbers are bypassed, resulting in uncontrolled release of radioactive iodine from the plant vent; or (3) Auxiliary Building temperatures are not controlled, resulting in failure of accident mitigating equipment.

By requiring OPERABILITY of all ABVS supply and exhaust fans, the proposed changes contained in this submittal assures Auxiliary Building negative pressure is maintained under all postulated post-accident, single-failure scenarios. The proposed changes to ABVS will not affect the elemental iodine adsorption capability of the system. Finally, engineering analyses conclude that these fan combinations, with single-active failures of the fans or their support systems considered, provide sufficient Auxiliary Building ventilation. Under the most limiting temperature conditions, the fans will maintain room temperatures within design limits. Accordingly, the consequences of a design basis LOCA, hence applicable design basis accidents or operational transients, are not increased.

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

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

ABVS supply fans are not considered essential to the primary safety-function of preventing or mitigating radioactive releases, nor are they currently required to be OPERABLE. Similarly, accident analyses take no credit for operation of supply fans. Accordingly, malfunctions of vital buses and ABVS exhaust fans are the only malfunctions of active ABVS related equipment important to safety that are previously evaluated.

The probability of failure of a vital bus is not increased by this proposal since the proposal has no direct effect on electrical power. Neither is the probability of exhaust fan failure increased by the proposal, since exhaust fans are not affected by this proposal, except that the number that must be OPERABLE is increased from two to three.

By requiring additional supply fans and exhaust fans to be OPERABLE, no single failure of either a vital bus or ABVS fan prevents (1) maintenance of negative Auxiliary Building pressure or (2) maintenance of temperatures within design limits. Since ABVS supply and exhaust fans cannot initiate accidents, increasing the number of fans required to be OPERABLE cannot create the possibility of a new or different kind of accident from any accident previously evaluated. In addition, the proposed changes to the ABVS surveillance testing concern ABVS leakage, HEPA filter and carbon adsorber capabilities, and laboratory test methods. Therefore, the proposed surveillance requirement changes would have no impact on the initiation of accidents.

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

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

The margin of safety is dependent upon the maintenance of specific operating parameters within designated design limits. Since iodine removal capability is not affected by the proposed changes, and negative Auxiliary Building pressure and temperatures will continue to be maintained within existing design limits under post-accident conditions, including consideration of the most limiting single active failure, the margin of safety is not reduced. By imposing new restrictions on the allowed outage times of ABVS components, the margin of safety is increased with the proposed changes to the ABVS Technical Specification Limiting Condition for Operation (LCO).

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

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

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

NRC Project Director: 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: November 4, 1997.

Description of amendment request: The amendments would change Technical Specification (TS) 3/4.6.2, ''Containment Spray System,'' to verify on recirculation flow that the containment spray pumps develop a differential pressure of at least 204 psi.

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 change revises the CS [containment spray] pump technical specification surveillance test acceptance from pump discharge pressure to pump differential pressure. This will account for the effect of RWST [refueling water storage tank] level on test results and provide acceptance criteria that verifies each CS pump performs as assumed in the accident analyses. This surveillance test is also being added to the Salem Unit 1 TS. The proposed change does not alter the physical plant arrangement or the method of CS pump inservice testing. Therefore it does not increase the probability of an accident. There is no change to pump performance requirements as assumed in the accident analyses. There is no change to CS system performance in response to an accident. Therefore, the proposed change does not involve an increase in the consequences of an accident previously evaluated.

The proposed change also corrects a typographical error by removing a repeated word. This change does not involve an 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.

The proposed change revises the Salem Unit 2 CS pump surveillance test acceptance criteria from pump discharge pressure to pump differential pressure. This will account for the effect of RWST level on test results and provide acceptance criteria that verify the CS pumps perform as assumed in the accident analyses. This surveillance test is also being added to the Salem Unit 1 TS. The proposed change does not alter the plant configuration. The change does not alter the method of performing inservice testing on the CS pumps. The change does not alter the CS pump performance assumed in the accident analyses. Therefore, the change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change also corrects a typographical error by removing a repeated word. This 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 proposed change ensures the CS pump Salem Unit 2 TS surveillance test acceptance criteria verify CS pump performance as assumed in the accident analyses accounting for RWST level effects. This surveillance test is also being added to the Salem Unit 1 TS. The proposal does not change the CS pump performance requirements assumed in the accident analyses and thus does not reduce the margin of safety.

The proposed change also corrects a typographical error by removing a repeated word. This does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: 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: November 14, 1997.

Description of amendment request: The proposed changes to the Technical Specifications (TSs) include administrative and editorial changes to correct errors in the TSs that have either existed since initial issuance or were introduced during subsequent changes. In addition, surveillance requirements are added that are considered administrative changes since the surveillances should have been incorporated with the TS when the applicable amendment to the TSs was approved by the NRC.

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

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

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

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

The proposed changes are administrative and editorial corrections to the TS that do not affect the ability of plant SSCs to perform their design basis accident functions. There is no reduction in the current surveillance requirements required to demonstrate the operability of plant SSCs. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: 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: November 14, 1997.

Description of amendment request: The proposed changes to the Technical Specifications (TSs) include administrative and editorial changes to correct errors in the TSs that have either existed since initial issuance or were introduced during subsequent changes. In addition, surveillance requirements are added that are considered administrative changes since the surveillances should have been incorporated with the TS when the applicable amendment to the TSs was approved by the NRC.

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

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

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

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

The proposed changes are administrative and editorial corrections to the TS that do not affect the ability of plant SSCs to perform their design basis accident functions. There is no reduction in the current surveillance requirements required to demonstrate the operability of plant SSCs. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John F. Stolz.

Public Service Electric & Gas Company, Docket No. 50–311, Salem Nuclear Generating Station, Unit No. 2, Salem County, New Jersey

Date of amendment request: October 29, 1997.

Description of amendment request: The proposed amendment would make a one-time change to Technical Specification 3/4.4.6, ''Steam Generators,'' to require that the next inspection be performed within 24 months of criticality for fuel cycle 10, rather than within 24 months from the previous inspection. The previous inspection was performed in May 1996; thus, adhering to the current Technical Specification would require inspection by May 1998 and would require a forced outage. It would also eliminate description of an alternate sampling plan that was applicable only to Unit 2's fourth refueling outage.

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 Design Basis Accident (DBA) analyzed in UFSAR Chapter 15.4.4, is Steam Generator Tube Rupture. The Technical Specification steam generator tube inspection attempts to avoid this DBA by maintenance of the integrity of the primary to secondary coolant boundary represented by steam generator tubes. The process by which this integrity is maintained is inspection of steam generator tubes at prescribed intervals, and the removal of defective tubes from service. Inspection intervals are based on preventing corrosion growth from exceeding tube structural strength, thereby preventing tube failure. An extensive steam generator inspection in May of 1996 characterized existing steam generator tube degradation, and degraded tubes were removed from service at that time. Degradation growth rates were evaluated for the next operating interval and it was determined that full cycle operation would not challenge tube structural integrity. Because degraded tubes were plugged, the integrity of the steam generators has been restored, and, because further degradation was prevented by a strictly controlled wet lay-up program in place since the inspection, steam generator integrity has since been maintained at the May 1996 level. This is the level normally expected for commencement of full power operations at the beginning of a fuel cycle. Thus, it can be reasonably concluded that this request to extend the inspection interval to conclude 24 months after the start of Unit 2 fuel cycle 10 does not involve an increase in the probability of an accident previously analyzed.

Salem UFSAR Chapter 15, Section 15.4.4., discusses the Design Basis Accident involving steam generator tube rupture. Since the Salem Unit 2 steam generators were extensively inspected and all degraded tubes were removed from service by plugging, integrity of the generators was restored to fully serviceable condition at that time. Degradation of steam generator tubes has been prevented since the inspection by a carefully controlled, EPRI Guidelines based, corrosion prevention program. It follows, then, that the Unit 2 steam generators were in the same condition immediately prior to fill and vent as if the inspection had just been concluded. This is the condition assumed for commencement of normal operation. Thus, it is reasonable to conclude that this proposal to extend the current steam generator inspection interval to end 24 months after start of Unit 2 fuel cycle 10 represents no significant increase in the consequences of an accident previously analyzed.

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

Steam generator tube inspections determine tube integrity and provide reasonable assurance that a tube rupture or primary to secondary leak will not occur. Accidents involving steam generator tube rupture are analyzed in Salem UFSAR Section 15.4.4, Steam Generator Tube Rupture. The only type of accident that can be postulated from extending the steam generator inspection interval would be a tube leak or rupture. Thus, it can be concluded that extending the steam generator inspection interval on a one-time basis cannot create the possibility of a different kind of accident from any accident previously evaluated.

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

The margin of safety, as with any TS, depends upon maintenance of specific operating parameters within design limits. In the case of steam generators, that margin is maintained through assurance of tube integrity as the primary to secondary boundary. Assurance of tube integrity is provided through periodic inservice testing of tube integrity and removal from service of defective tubes. Additional margin is provided through protection from possible consequences of steam generator tube failure by detection and mitigation systems. As discussed in 1., above, there was an extensive steam generator inspection, and the steam generators have been maintained since the inspection, using a lay-up program that complies with EPRI Guidelines, to prevent further tube degradation. Also, N–16 monitors were added, enhancing detection capabilities. The margin as established by the latest inspection has been maintained by the corrosion control program of EPRI Primary and Secondary Guidelines based on wet lay-up conditions. Thus, it can be reasonably concluded that this proposal to amend the Salem Unit 2 Technical Specifications, on a one-time basis, to extend the steam generator inspection interval to end 24 months after start of Unit 2 fuel cycle 10 does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John F. Stolz.

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

Date of application request: August 8, 1997.

Description of amendment request: The proposed amendment would revise the surveillance requirements (SR) of Technical Specification (TS) 3/4.7.4 ''Essential Service Water System'' by removing the requirement to perform SR 4.7.4.b.1, 4.7.4.b.2 and 4.7.4.c during shutdown.

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 change to TS has no adverse impact on the probability of occurrence or the consequences of an accident. The proposed amendment does not change or alter the design assumptions for the systems or components used to mitigate the consequences of an accident and the methodologies used in the accident analysis remain unchanged. The operating limits and the radiological consequences will not be changed. No design basis accidents will be affected by this change since the required TS surveillances will continue to be performed on an 18 month frequency.

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

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

All design and performance criteria continue to be met and no new failure mechanisms have been identified. The proposed change does not affect the design or operation of any system or component in the plant since the required TS surveillances will continue to be performed on an 18 month frequency. The safety functions of the related structures, systems or components are not changed in any manner, nor is the reliability of any structure, system or component reduced. Conducting these surveillances online will not increase the possibility of plant transients. Since the safety functions and reliability are not adversely affected, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not affect or change a safety limit or affect plant operations since the required TS surveillances will continue to be performed on an 18 month frequency. This change will not reduce the margin of safety assumed in the accident analysis nor reduce any margin of safety as defined in the basis for any TS.

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: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Project Director: William H. Bateman.

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

Date of application request: August 8, 1997.

Description of amendment request: The proposed amendment would revise Table 3.3–3, Functional Units 4.b.2 and 5.a.2 of the Callaway Technical Specifications (TS) by (1) changing the main steam and feedwater isolation system (MSFIS) channels to be consistent with the requirements for the solid state protection system (SSPS), (2) adding a clarifying note, and (3) deleting and replacing Action Statements 27a and 34a with Action Statements 27 and 34. In addition, Table 4.3–2, Functional Units 4.b and 5.a are proposed to be revised by changing the slave relay quarterly surveillance to a quarterly actuation logic test for the MSFIS actuation and relays.

*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 to Technical Specifications (TS) have no adverse impact on the probability of occurrence or the consequences of an accident. The proposed amendment does not change or alter the design assumptions for the systems or components used to mitigate the consequences of an accident and the methodologies used in the accident analysis remain unchanged. The operating limits and the radiological consequences will not be changed. No design basis accidents will be affected by these changes. The proposed changes do not result in any hardware changes.

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

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

The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. All design and performance criteria continue to be met and no new failure mechanisms have been identified. The proposed changes do not affect the design or operation of any system or component in the plant. The safety functions of the related structures, systems or components are not changed in any manner, nor is the reliability of any structure, system or component reduced. However, these changes are consistent with the requirements for the SSPS. Since the safety functions and reliability are not adversely affected, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes will not affect or change a safety limit or affect plant operations. These changes will not reduce the margin of safety assumed in the accident analysis nor reduce any margin of safety as defined in the basis for any TS. The proposed changes do not affect the acceptance criteria for any analyzed event. No setpoints are revised and the system response time will not be affected.

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: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Project Director: William H. Bateman.

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

Date of application request: August 8, 1997.

Description of amendment request: The proposed amendment would revise Table 3.7–2 of the Technical Specifications to specify that the lift setting tolerance for the main steam line safety valves be +3/−1% as-found and plus or minus 1% as-left. Table 2.2–1 would be revised by reducing the sensor error for the pressurizer pressure-high trip.

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 main steam line safety valves are designed to mitigate transients by preventing overpressurization of the main steam system. The proposed change does not alter this design basis. The revised analysis shows that the probability or consequences of all previously analyzed accidents are not changed by increasing the setpoint tolerance of the safety valves. Therefore, there is no increase in the probability of occurrence or the consequences of any accident.

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

There is no new type of accident or malfunction created, the method and manner of plant operation will not change nor is there a change in the method in which any safety related system performs its function. Any main steam safety valve lifting at the extremes of the proposed tolerance will not result in a low lift setpoint that is less than the normal no load system pressure or a high lift setpoint that allows main steam system overpressurization.

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

This is based on the fact that no plant design changes are involved and the method and manner of plant operation remains the same. With the increased setpoint tolerance, the main steam safety valves will still prevent pressure from exceeding 110 percent of design pressure in accordance with the ASME code. All FSAR accident analysis conclusions remain valid and unaffected by this change.

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

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037

NRC Project Director: William H. Bateman.

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

Date of application request: August 8, 1997.

Description of amendment request: The proposed amendment application would revise feedwater isolation engineered safety feature actuation system (ESFAS) functions in Technical Specification Tables 3.3–3, 3.3–4 and 4.3–2 as follows:

(1) The Applicable MODES for Functional Units 5.a.1), Automatic Actuation Logic and Actuation Relays, and 5.a.2), Automatic Actuation Logic and Actuation Relays, in Tables 3.3–3 and 4.3–2 would be revised to add MODE 3.

(2) A new Functional Unit 5.d, Steam Generator (SG) Water Level Low-Low (for feedwater isolation only), would be added to Tables 3.3–3, 3.3–4, and 4.3–2.

(3) In conjunction with the changes under item (2), the Applicable MODES in Table 3.3–3 for AFW SG Water Level Low-Low Functional Units 6.d.1).c), Start Motor-Driven Pumps Vessel delta T (Power-1, Power-2), would be revised to delete MODE 3. Functional Unit 6.d.3) in Table 4.3–2 would also be revised to delete MODE 3.

(4) The Bases for Functional Unit 11.b, Reactor Trip P–4, in Table 3.3–3 would be revised to add a note allowing the feedwater isolation function on P–4 coincident with low T_{avg} to be blocked.

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:

Actuation Logic Applicability and New SG Water Level Low-Low Functional Unit

1. The proposed change does not involve a significant increase in the probability or

consequences of an accident previously evaluated.

The proposed changes impose more stringent requirements and have been reviewed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure the plant's operation and testing are consistent with the safety analysis and licensing basis. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed other than the bypass switch addressed in a separate 50.92 evaluation below) or changes in controlling parameters. The proposed changes do impose different requirements; however, these changes are consistent with assumptions made in the safety analysis and licensing basis. Actuation logic applicability is extended to MODE 3 and the SSPS slave relays that implement feedwater isolation on SG water level low-low will continue to be surveilled quarterly as they have always been tested. Thus, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The imposition of more stringent requirements does not reduce the margin of safety. The margin of safety would be increased since the scope of the Technical Specifications has been increased to include additional plant equipment and add additional Applicability requirements. The changes are consistent with the safety analysis and licensing basis. Therefore, the proposed changes do not involve a reduction in a margin of safety.

TTD Applicability

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

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The proposed change adds a relaxation to the Applicability for the SG Water Level Low-Low Vessel delta T channels. The proposed change in the Applicability will not affect any of the analysis assumptions for any of the accidents previously evaluated. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. A Vessel delta T channel should only be tripped if it is inoperable and the reactor is operating, when the need to restrict trip time delays is applicable. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Accident analyses have been performed with the maximum trip time delays enabled at power levels up to 19% RTP (10% RTP plus uncertainty). Therefore, operation in MODE 3 with the maximum trip time delays is enveloped. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change in Applicability will not impact the normal method of plant operation. The maximum trip time delay should be enabled in MODE 3 to preclude an unnecessary feedwater isolation or auxiliary feedwater actuation from occurring prior to the expiration of the trip time delay previously analyzed for MODE 1 operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new of different kind of accident from any previously evaluated.

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

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

Feedwater Isolation on P–4/Low T_{avg} Bypass Switch

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

Overall protection system performance will remain within the bounds of the previously performed accident analyses. The P–4/Low T_{avg} Bypass Switch design change will not impact any accidents previously evaluated in the FSAR since feedwater isolation upon reaching this function was never credited.

The ESFAS will continue to function in a manner consistent with the accident analysis assumptions and the plant design basis. As such, there will be no degradation in the performance of nor an increase in the number of challenges to equipment assumed to function during an accident situation.

This Technical Specification change does not affect the probability of any event initiators. There will be no change to normal plant operating parameters or accident mitigation capabilities. Therefore, there will be no increase in the probability or consequences of any accident occurring due to this change.

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

There are no changes in the method by which any safety-related plant system performs its safety function and the normal manner of plant operation is unaffected, other than the proposed allowance to bypass feedwater isolation on P–4 coincident with low T_{avg}. This bypass switch modification will be performed under the design standards applicable to all safety system bypasses at Callaway, except for Section 4.12 of IEEE 279–1971. Section 4.12 of IEEE 279–1971 requires that an operating bypass of a protective function be automatically removed whenever permissive conditions are not met. However, the subject circuitry does not provide a protective function. It is not assumed or credited in any safety analysis. In addition, plant conditions that would call for the restoration of the feedwater isolation function cannot occur without operator action to close the reactor trip breakers. Administrative controls will govern the proper use of and restoration from the proposed bypass. Although the addition of the bypass switch introduces the potential for an equipment malfunction of a different type from any previously evaluated in the FSAR, the possibility of a new or different type of accident is not created. The switch functions only to allow a manual bypass of feedwater isolation. The failure of the switch or its improper use will not be an event initiator for the previously analyzed Loss of Normal Feedwater event in FSAR Section 15.2.7 since it cannot fail in such a manner as to cause feedwater isolation.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. There will be no adverse effect or challenges imposed on any safety-related system as a result of this change. Therefore, the possibility of a new or different type of accident is not created.

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

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on DNBR limits, F_Q, F-delta-H, LOCA PCT, peak local power density, or any other 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: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Project Director: William H. Bateman.

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

Date of amendment request: November 5, 1997.

Description of amendment request: The current Technical Specifications requirements prohibit loads in excess of 2500 pounds from traveling over irradiated fuel assemblies in the spent fuel pit. Due to the number of irradiated fuel assemblies currently stored in the spent fuel pit over years of operation, additional flexibility is needed to accomplish the movement of the spent fuel pit gates during refueling activities and to reduce fuel handling activities in preparation for refueling outages. In order to perform gate seal maintenance prior to each outage, a gate is moved across the irradiated fuel storage area to the cask handling area where it can be lifted out of the spent fuel pit. When a clear path of empty fuel storage cells cannot be established, seal maintenance cannot be performed unless relief from the current Limiting Condition of Operation is granted. The proposed changes will exempt these requirements for the movements of the spent fuel gates provided specific administrative controls are satisfied.

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:

Specifically, operation of the North Anna Power Station in accordance with the proposed changes will not:

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

The accident in question is a fuel handling accident in the spent fuel pit. The proposed changes will actually reduce the probability of a fuel handling accident by eliminating unnecessary fuel assembly movements. After this change is implemented, only those assemblies containing control rod assemblies will be subjected to such moves prior to movement of the gates instead of the current practice of moving all the fuel necessary to establish a load path of empty cells. A redundant rigging system will be provided which eliminates the possibility of a load drop due to a hoist failure. Furthermore, even though the double rigging system makes a load drop due to a hoist failure an incredible event, a calculation was performed to determine the effects of a direct impact load on a single fuel storage cell or the SFP [spent fuel pit] structure. The calculation concludes that there will be no adverse consequences to either irradiated fuel or the SFP structure. The plant design basis fuel handling accident will not be violated. Therefore, with the administrative controls in place to eliminate the possibility of a gate drop the probability of occurrence or the consequences of a fuel handling accident are not increased.

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

The proposed changes establish adequate administrative controls over the spent fuel pit gate movements to prevent damage to stored irradiated fuel and fuel racks thereby ensuring the design basis fuel handling accident remains bounding and that fuel spacing is maintained in the racks precluding criticality.

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

The new administrative controls ensure that a postulated gate drop will not occur due to compliance with our licensing commitments to NUREG–0612 and the requirement to install a redundant rigging system to eliminate the possibility of a load drop initiated by hoist failure. Analysis has determined that in the event the gate was to be dropped from its controlled lift height: (1) There will be no damage to irradiated fuel caused by the direct impact loading on a single storage cell and (2) the fuel storage rack will maintain fuel in a non-critical array. A new criteria, demonstrating the ability of the pool floor to remain intact after a gate drop has been shown by analysis. New controls prevent the degradation of the existing margin of safety and ensure an adequate safety margin for the new criteria. The administrative controls added for the gate lift preclude the possibility of a load drop induced by a hoist failure and, therefore ensure the potential for radioactivity release and inadvertent criticality remain bounded by the present design basis. Therefore, the margin of safety is not reduced by the proposed change.

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

Local Public Document Room location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903–2498.

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Project Director: James E. Lyons.

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

Date of amendment request: November 18, 1997.

Description of amendment request: The Technical Specifications surveillance requirements currently require testing and inspection of the Turbine Overspeed Protection System control valves, at least once per 31 days, to ensure their ability to prevent overspeeding of the turbine. Based on an analysis of Westinghouse BB–296 turbines with steam chests, the proposed change would increase the surveillance test interval from at least once per 31 days to at least once per 92 days.

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:

Specifically, operation of the North Anna Power Station in accordance with the proposed Technical Specifications changes will not:

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

No new or unique accident precursors are introduced by these changes in surveillance requirements. The probability of turbine missile ejection with an extended test interval to 92 days for the turbine governor and throttle valves has been determined to remain within the applicable NRC acceptance criteria. The heavy hub design of the turbine rotors provides further assurance that the probability of ejection of turbine missiles due to destructive overspeed remains within the acceptance criteria. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The demonstrated high reliability of the turbine governor and throttle valves and the verification of the operability of the other turbine control valves provide adequate assurance that the turbine overspeed protection system will operate as designed, if needed. Turbine governor and throttle valve testing performed to date has demonstrated the reliability of these valves. In addition, the operability of the other turbine valves (i.e., reheat and intercept stop valves) will continue to be verified every 18 months as required by the Technical Specifications.

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

Since the implementation of the proposed change to the surveillance requirements will not require hardware modifications (i.e., alterations to plant configuration), operation of the facilities with these proposed Technical Specifications does not create the possibility for any new or different kind of accident which has not been already been evaluated in the Updated Final Safety Analysis Report (UFSAR). In addition, the results of the probabilistic evaluation indicate that no additional transients have been introduced.

The proposed revision to the Technical Specifications will not result in any physical alteration to any plant system, nor would there be a change in the method by which any safety-related system performs its function. The design and operation of the turbine overspeed protection and turbine control systems are not being changed.

The proposed Technical Specifications changes do not affect the design, operation, or failure modes of the valves and other components of the turbine overspeed protection system. Therefore, the proposed change does 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 do not reduce the margin of safety as defined in the basis for any Technical Specifications. Furthermore, the total turbine missile ejection probability continues to be enveloped by the applicable acceptance criteria of 1E–5. The design and operation of the turbine overspeed protection and turbine control systems are not being changed and the operability of the turbine governor and throttle valves will be demonstrated on a refuelling outage basis. In addition, the results of the accident analyses, which are documented in the UFSAR, continue to bound operation with the proposed change in surveillance interval for the turbine throttle and governor valves, so that there is no safety margin reduction. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903–2498.

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Project Director: James E. Lyons.

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.

Tennessee Valley Authority, Docket Nos. 50–327, Sequoyah Nuclear Plant, Unit 1, Hamilton County, Tennessee

Date of application for amendments: November 21, 1997.

Description of amendments request: Amend Technical Specifications to add a one-time allowance through Operating Cycle 9 to Surveillance Requirement 4.4.3.2.1.b to perform stroke testing of the power-operated relief valve in Mode 5 rather than in Mode 4.

Date of publication of individual notice in the **Federal Register**: December 1, 1997 (62 FR 63565).

Expiration date of individual notice: December 31, 1997.

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

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 Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of application for amendments: November 30, 1995, as supplemented March 15, 1996, March 6, 1997, and June 27, 1997.

Brief description of amendments: The amendments incorporate references to a new Combustion Engineering, Inc. topical report describing steam generator tube sleeves, delete references to the previous CE topical report, incorporate sleeve/tube inspection scope and expansion criterion, revise the plugging limit for a CE sleeve to 28% of the nominal sleeve wall thickness, and incorporate a post weld heat treatment for free span welds.

Date of issuance: November 18, 1997.

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

Amendment Nos.: 223 and 199.

Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 3, 1996 (61 FR 176). The March 15, 1996, March 6, 1997, and June 27, 1997, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated November 18, 1997.

No significant hazards consideration comments received: No.

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

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 application for amendments: August 6, 1997.

Brief description of amendments: The amendments address an unreviewed safety question associated with the handling of the spent fuel shipping cask at the Brunswick Steam Electric Plant, Units 1 and 2.

Date of issuance: December 2, 1997.

Effective date: December 2, 1997.

Amendment Nos.: 190 and 221.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments authorize changes to the facility's Updated Final Safety Analysis Report.

Date of initial notice in **Federal Register**: September 17, 1997 (62 FR 48897) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 2, 1997.

No significant hazards consideration comments received: No.

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

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of application for amendment: February 18, 1997.

Brief description of amendment: This amendment revises the maximum allowable power range neutron flux high setpoints (percent of rated thermal power) shown in Technical Specification Table 3.7–1.

Date of issuance: November 25, 1997.

Effective date: November 25, 1997.

Amendment No.: 75.

Facility Operating License No. NPF–63: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: April 9, 1997 (62 FR 17225) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 25, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

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

Date of application for amendments: April 7, 1997, as supplemented on August 7, 1997.

Brief description of amendments: The amendments revise the technical specifications to permit installation and use of C&D Charter Power Systems, Inc., batteries.

Date of issuance: November 25, 1997.

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

Amendment Nos.: 93 and 93.

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

Date of initial notice in **Federal Register**: October 22, 1997 (62 FR 54868). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 25, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: October 13, 1997, as supplemented by letters dated October 28 and November 5, 1997.

Brief description of amendments: The amendments revise TS Table 3.3–4, ''Engineered Safety Features [ESF] Actuation System Instrument Trip Setpoints.'' Specifically, the amendments support the replacement of three safety-related narrow range Refueling Water Storage Tank level instruments with three safety-related wide range level instruments. The ESF trip setpoint for the refueling water automatic switchover to recirculation is revised to account for the difference in instrument uncertainty associated with wide range level instruments and provides additional operator response time margin.

Date of issuance: November 25, 1997.

Effective date: Unit 1—As of the date of issuance to be implemented consistent with the refueling outage scheduled for June 1998; Unit 2—As of the date of issuance to be implemented within 30 days from the date of issuance.

Amendment Nos.: 177 (Unit 1); 159 (Unit 2).

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

Date of initial notice in **Federal Register**: October 22, 1997 (62 FR 54859). The October 28 and November 5, 1997, letters provided additional and clarifying information that did not change the scope of the October 13, 1997, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 25, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North Carolina.

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

Date of application for amendments: October 10, 1997, as supplemented by letters dated November 3, 6, and 10, 1997.

Brief description of amendments: The amendments revise Technical Specifications to implement alternate repair criteria for steam generator tubes that have degraded roll joints inside of the upper tubesheet. The alternate repair criteria would allow new roll joints to be installed below the degraded roll joints in the upper tubesheet.

Date of issuance: November 21, 1997.

Effective date: November 21, 1997.

Amendment Nos.: Unit 1—227; Unit 2—227; Unit 3—224.

Facility Operating License Nos. DPR–38, DPR–47, AND DPR–55: The amendments revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: Yes. (62 FR 55835 dated October 28, 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 November 28, 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 amendments.

The Commission's related evaluation of the amendments, finding of exigent circumstances, and a final determination of no significant hazards consideration are contained in a Safety Evaluation dated November 21, 1997.

Attorney for licensee: M. J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW., Washington, DC.

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

Duquesne Light Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of application for amendment: March 10, 1997, as supplemented July 28 and September 17, 1997.

Brief description of amendment: The amendment modifies Technical Specification 3/4.4.5, ''Steam Generators,'' and its associated Bases and adds a new license condition to Appendix C for Beaver Valley Power Station, Unit No. 1 (BVPS–1) to allow repair of steam generator tubes by installation of sleeves developed by ABB Combustion Engineering. In addition, the amendment deletes the option for using the kinetic sleeving methodology previously approved for use at BVPS–1.

Date of issuance: November 25, 1997.

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

Amendment No.: 208.

Facility Operating License No. DPR–66: Amendment revised the Technical Specifications and License.

Date of initial notice in **Federal Register**: April 23, 1997 (62 FR 19829). The July 28 and September 17, 1997, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the April 23, 1997, **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 25, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001.

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

Date of application for amendment: September 12, 1997, as supplemeneted November 7, 1997.

Brief description of amendment: The proposed amendment involves a revision to the Emergency Diesel Generator protective relaying scheme at Crystal River Unit 3, to be reflected in the next revision to the Final Safety Analysis Report (FSAR).

Date of issuance: December 1, 1997.

Effective date: Effective upon issuance.

Amendment No.: 159.

Facility Operating License No. DPR–72:. Amendment revises the FSAR.

Date of initial notice in **Federal Register**: September 30, 1997 (62 FR 51165). By letter dated November 7, 1997, the licensee provided additional information which did not affect the original no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 1, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 32629.

GPU Nuclear Corporation, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of application for amendment: October 10, 1996, as supplemented March 25, June 6, and August 29, 1997.

Brief description of amendment: The amendment extends the instrumentation surveillances for the condenser low vacuum, high temperature main steamline tunnel, recirculation flow, and reactor coolant leakage. Additionally, the change extends the equipment test/operability checks for containment vent and purge isolation, electromagnetic relief valve operability, and drywell to torus leakage test.

Date of Issuance: November 26, 1997.

Effective date: November 26, 1997, with full implementation within 60 days.

Amendment No.: 193.

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

Date of initial notice in **Federal Register**: November 6, 1996 (61 FR 57485). The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated November 26, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 20, 1997, as supplemented on September 23, 1997.

Brief description of amendment: The amendment changes the Technical Specifications (TSs) by relocating the containment isolation valve (CIV) list from the TSs to the Technical Requirements Manual in accordance with Generic Letter 91–08, ''Removal of Component Lists from the Technical Specifications.'' The amendment also changes the surveillance requirement for valves, blind flanges, and deactivated automatic valves located inside containment that are locked, sealed, or otherwise secured in the closed position from once every 31 days to during each cold shutdown, but no more than once per 92 days. The TS Bases is changed to reflect the relocation of the containment isolation valve list from the TSs to the Technical Requirements Manual and dicusses administrative controls for CIV operation in Modes 1 through 4. Also, a license condition has been added to paragraph 2.C. of the Operating License to ensure enforceability and to provide a method of tracking the license condition back to the license amendment.

Date of issuance: November 19, 1997.

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

Amendment No: 210

Facility Operating License No. DPR–65: Amendment revised the Technical Specifications and License Conditions.

Date of initial notice in **Federal Register**: June 18, 1997 (62 FR 33128). The September 23, 1997, letter provided clarification of the initial submittal and did not affect the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 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 and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

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

Date of application for amendment: September 16, 1997.

Brief description of amendment: The amendment changes the main steam line American Society of Mechanical Engineers Code (Code) safety valves Technical Specifications (TSs) by: (1) Deleting TS Table 3.7.1, ''Maximum Allowable Power Level-High Trip Setpoint with Inoperable Steam Line Safety Valves During Operation with Both Steam Generators,'' by not allowing operation in Mode 1 or 2 with inoperable Code safety valves while allowing operation in Mode 3 with up to three Code safety valves inoperable per steam generator, (2) modifing the associated action statement in TS 3.7.1.1 to reflect the operational changes, and (3) updating the TS Bases to reflect the proposed changes and include the correct amendment history numbers to reflect previously approved amendments.

Date of issuance: November 19, 1997.

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

Amendment No.: 211.

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

Date of initial notice in **Federal Register**: October 8, 1997 (62 FR 52582). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 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, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

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: September 26, 1997.

Brief description of amendments: The amendments revise Technical Specification (TS) 3.4.B, ''Auxiliary Feedwater System,'' to provide specific guidance for conducting post-maintenance operational testing of the turbine-driven auxiliary feedwater pump and associated system valves to meet operability requirements and limiting conditions for operation during unit startup. Additionally, the amendments revise Table TS.3.5.2B to allow the auxiliary feedwater pump auto-start actuation instrumentation to be bypassed during startup and shutdown operations when the main feedwater pumps are not required to supply feedwater to the steam generators.

Date of issuance: November 25, 1997.

Effective date: November 25, 1997, with full implementation within 30 days.

Amendment Nos.: 134 and 126.

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

Date of initial notice in **Federal Register**: October 22, 1997 (62 FR 54874). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 25, 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: May 20, 1996.

Brief description of amendment: The amendment revises the technical specifications to correct and clarify surveillance test requirements for the reactor protective system and other plant instrumentation and control systems.

Date of issuance: November 24, 1997.

Effective date: November 24, 1997, to be implemented within 120 days of the date of issuance.

Amendment No.: 182.

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

Date of initial notice in **Federal Register**: August 28, 1996 (61 FR 44361). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 24, 1997.

No significant hazards consideration comments received: No.

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

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

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

Brief description of amendment: The amendment incorporates into the Technical Specifications the Margin Recovery portion of the Fuel Upgrade Margin Recovery Program and support increased steam generator plugging, improved fuel reliability, reduced fuel costs, longer fuel cycles, reduced spent fuel pool storage, and enhanced reactor safety.

Date of issuance: November 26, 1997.

Effective date: As of date of issuance. To be implemented on Unit 1 prior to entry into Mode 2 from the current outage.

Amendment No.: 201.

Facility Operating License No. DPR–70: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 3, 1996 (61 FR 34898). The March 19 and August 29, 1997, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 26, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: January 4, 1996.

Brief description of amendments: These amendments delete License Condition 2.C(26) for SONGS Unit 2 and License Condition 2.C(27) for SONGS 3. These license conditions require that Southern California Edison implement and maintain a plan for scheduling all capital modifications based on an NRC approved Integrated Implementation Schedule Program Plan.

Date of issuance: December 3, 1997.

Effective date: December 3, 1997.

Amendment Nos.: Unit 2—137; Unit 3—129.

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

Date of initial notice in **Federal Register**: April 10, 1996 (61 FR 15997). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 3, 1997.

No significant hazards consideration comments received: No.

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

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 application for amendment: January 20, 1997.

Brief description of amendment: This amendment revises Technical Specification (TS) Section 3/4.5.2, ''Emergency Core Cooling Systems, ECCS Subsystems – T_{avg} greater than or equal to 280°F,'' TS Section 3/4.5.3, ''Emergency Core Cooling Systems, ECCS Subsystems – T_{avg} less than 280 °F,'' and TS Section 3/4.7, ''Plant Systems.'' Several surveillance intervals were changed from 18 months to once each refueling interval.

Date of issuance: December 2, 1997.

Effective date: December 2, 1997.

Amendment No.: 216

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

Date of initial notice in **Federal Register**: March 12, 1997 (62 FR 11498). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 2, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, Ohio 43606.

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 application for amendment: September 17, 1996, as supplemented by letters dated November 27, 1996, and October 14, 1997.

Brief description of amendment: This amendment revises the surveillance interval from 18 months to less than or equal to 730 days, nominally 24 months, for Technical Specification (TS) 3/4.5.2, ''Emergency Core Cooling Systems—ECCS Subsystems—T_{avg} greater than or equal to 280 degrees F''; TS 3/4.6.5.1, ''Containment Systems—Shield Building—Emergency Ventilation System''; TS 3/4.7.6.1, ''Plant Systems—Control Room Emergency Ventilation System''; TS 3/4.7.7, ''Plant Systems—Snubbers''; TS 3/4.9.12, ''Refueling Operations—Storage Pool Ventilation''; and TS Bases 3/4.7.7—''Snubbers.''

Date of issuance: December 2, 1997.

Effective date: Immediately, and shall be implemented no later than 120 days after issuance.

Amendment No.: 217.

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

Date of initial notice in **Federal Register**: October 9, 1996 (61 FR 52972). The supplemental information submitted by the licensees did not impact the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 2, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, Ohio 43606.

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

Date of application for amendment: December 11, 1996 (as supplemented by letter dated January 6, 1997), January 30, 1997 (as supplemented by letter dated September 15, 1997), and April 18, 1997.

Brief description of amendment: This amendment extends surveillance requirement intervals from 18 to 24 months, revises setpoints, and revises TS 2.2, ''Limiting Safety System Settings.'' Administrative changes have also been made.

Date of issuance: December 2, 1997.

Effective date: December 2, 1997.

Amendment No.: 218.

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

Dates of initial notice in **Federal Register**: January 15, 1997 (62 FR 2194), March 12, 1997 (62 FR 11498) and June 4, 1997 (62 FR 30654). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 2, 1997.

No significant hazards consideration comments received: No. The supplemental information provided by the licensees did not affect the proposed no significant hazards consideration.

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, OH 43606.

Dated at Rockville, Maryland, this 10th day of December 1997.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 97–32763 Filed 12–16–97; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

[Docket Nos. 50–315 and 50–316]

American Electric Power Company; Receipt of Petition for Director's Decision Under 10 CFR 2.206

Notice is hereby given that by Petition dated October 9, 1997, David A. Lochbaum, on behalf of the Union of Concerned Scientists, has requested that the U.S. Nuclear Regulatory Commission (NRC) take action with regard to Donald C. Cook Nuclear Plant, Units 1 and 2, operated by American Electric Power Company (the Licensee). The Petition requests that the operating licenses for D.C. Cook, Units 1 and 2, be modified, revoked, or suspended until there is reasonable assurance that the Licensee's systems are in conformance with design-and licensing-bases requirements. The Petition requests that systems with a safety function at D.C. Cook be qualified and capable of performing their required function under all design conditions before restart. The Petition also requests that a public hearing into this matter be held in the Washington, DC, area before the first unit at D.C. Cook is authorized to restart.

As the basis for these requests, the Petition states that the NRC recently completed an architect/engineer design inspection at D.C. Cook. The Licensee had previously reviewed the same systems as part of its design-basis documentation reconstitution program. Findings by the NRC inspection team led to a shutdown of both units and has necessitated changes to the plant's physical configuration. Therefore, the Petition asserts that the Licensee's design-basis documentation reconstitution and updated final safety analysis report validation programs lack the necessary rigor and focus. The Petition further asserts that deficiencies in the Licensee's design control programs may also be responsible for similar problems in its safety systems, which were not examined by the NRC.

The request is being treated pursuant to 10 CFR 2.206 of the Commission's regulations. The request has been referred to the Director of the Office of Nuclear Reactor Regulation. As provided by 10 CFR 2.206, appropriate action will be taken on this Petition within a reasonable time. A copy of the Petition is available for public inspection at the Commission's Public Document Room, located at the Gelman Building, 2120 L Street, NW., Washington, DC 20555–0001.

Dated at Rockville, Maryland, this 9th day of December 1997.

For The Nuclear Regulatory Commission.

Samuel J. Collins,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 97–32878 Filed 12–16–97; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request

Upon Written Request, Copies Available From: Securities and Exchange