These time limits may be adjusted depending on the number of presentations and comment. The workshop will be transcribed, and the transcript will be available at the NRC Public Document Room.

To foster meaningful discussions during this session and to aid participants in preparing their presentations and comments, participants should consider the following set of questions:

- What impact will the CBLA Administrative Letter have on those organizations that the NRC regulates?
- Should the NRC develop a CBLA database that could be made available to the public?
- What are the reasons that the CBLA program has not been used more widely by licensees?
- What are the savings that can result from conversion to the improved Standard Technical Specifications?

Dated In Rockville, Maryland, this 9th day of March , 1995.

For the Nuclear Regulatory Commission.

Eugene V. Imbro,

Director, RRG/CBLA Programs, Office of Nuclear Reactor Regulation.

[FR Doc. 95–6341 Filed 3–14–95; 8:45 am]

BILLING CODE 7590–01–M

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from February 16, 1995, through March 3, 1995. The last biweekly notice was published on March 1, 1995.

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

By April 14, 1995, 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, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to **(Project Director)**: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, 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.

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 amendments request: January 31, 1995

Description of amendments request: The proposed amendments would revise the Technical Specifications (TSs) for Calvert Cliffs, Unit Nos. 1 and 2, to increase the amount of Trisodium Phosphate Dodecahydrate (TSP) located in the containment sump baskets required to be verified by TS surveillance. The requested change is the result of an reanalysis of the amount of TSP necessary to maintain the appropriate pH in the containment sump water subsequent to a Loss of Coolant Accident. Specifically, the request would change the TS value of TS 4.5.2.e.3 from the existing amount of 100 ft^3 to 289 ft^3. TS 4.5.2.e.4 would also be changed by moving the amounts of TSP and refueling water storage tank water to be used in the required tests to the TS Bases Section 3/4.5.2 and 3/4.5.3. These Bases sections would also be changed by modifying the test methods.

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. Would not involve a significant increase in the probability orconsequences of an accident previously evaluated.

Trisodium Phosphate Dodecahydrate (TSP) is stored in the containment lower level to raise the pH of the sump and spray water following a Loss of Coolant Accident (LOCA). As the pH of the water increases, more radioactive iodine is kept in solution and the possibility of airborne radioactivity leakage is decreased. An additional advantage of a higher pH is the beneficial reduction in chloride stress corrosion cracking of metal components in the containment following an accident.

This chemical is an accident mitigator, not an accident initiator in that it is not used until after an accident has occurred. At the time it goes into solution, the accident has occurred, containment spray has been activated and water has collected in the containment sump. Therefore, increasing the Technical Specification minimum amount verified to be in each containment will not involve a significant increase in the probability of an accident previously evaluated.

Updated Final Safety Analysis Report, Chapter 14.24, ''Maximum Hypothetical Accident'', uses an assumption of a pre-RAS minimum containment spray pH of 5.0 for the iodine removal calculation and a post-RAS sump pH of 7.0 for iodine retention. Raising the pH to 7.0 does not increase the consequences of an accident previously evaluated.

The proposed change to Technical Specification 4.5.2.e.4 would remove the amounts of chemical and water used in the test to the Bases. This relocation will not alter the test method or acceptance criteria, but will allow adjustments to the ratio of TSP and borated water under the controls of 10 CFR 50.59 to reflect changes in plant conditions. In the Bases, the amount of TSP used in the test is changed to reflect the ratio of TSP to water that would be found in the containment following a LOCA. The specified concentration of boron in the test reflects the highest concentration that could be found in the containment following a LOCA. The test temperature is changed to 120°F which is well below the temperature expected to be found in the containment sump following a LOCA. The decanting of the solution does not change the intent of the test method since the dissolving period will still be conducted without agitation.

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

2. Would not create the possibility of a new or different type of accident from any accident previously evaluated.

The addition of more TSP does not represent a significant change in the configuration or operation of the plant. Trisodium Phosphate Dodecahydrate is currently present in the containment lower level. There are no physical changes which result from the increase in volume. The proposed change to Technical Specification 4.5.2.e.4 would move the amounts of chemical and water used in the test to the Bases. This relocation will not alter the test method or acceptance criteria, but will allow adjustments to the ratio of TSP and borated water under the controls of 10 CFR 50.59 to reflect changes in plant conditions. In the Bases, the amount of TSP used in the test is changed to reflect the ratio of TSP to water that would be found in the containment following a LOCA. The specified concentration of boron in the test reflects the highest concentration that could be found in the containment following a LOCA. The test temperature is changed to 120°F which is well below the temperature expected to be found in the containment sump following a LOCA. The decanting of the solution does not change the intent of the test method since the dissolving period will still be conducted without agitation.

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

3. Would not involve a significant reduction in a margin of safety.

Trisodium Phosphate Dodecahydrate is stored in the containment lower level to raise the pH of the sump and spray water following a LOCA. As the pH of the water increases, more radioactive iodine is kept in solution and the possibility of airborne radioactivity leakage is decreased. Additionally, a higher pH has a beneficial effect on chloride stress corrosion cracking of metal components in the containment.

Technical Specification 4.5.2.e.3 requires verification that a minimum volume of TSP is contained in the storage baskets in each containment. This change proposes to increase that volume. The increased volume will ensure the containment sump, when filled with water, will have an acceptable pH following a LOCA.

The proposed change to Technical Specification 4.5.2.e.4 would move the amounts of chemical and water used in the test to the Bases. This relocation will not alter the test method or acceptance criteria, but will allow adjustments to the ratio of TSP and borated water under the controls of 10 CFR 50.59 to reflect changes in plant conditions. In the Bases, the amount of TSP used in the test is changed to reflect the ratio of TSP to water that would be found in the containment following a LOCA. The specified concentration of boron in the test reflects the highest concentration that could be found in the containment following a LOCA. The test temperature is changed to 120°F which is well below the temperature expected to be found in the containment sump following a LOCA. The decanting of the solution does not change the intent of the test method since the dissolving period will still be conducted without agitation.

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

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

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

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

NRC Project Director: Ledyard B. Marsh

Boston Edison Company, Docket No. 50-293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: February 9, 1995

Description of amendment request: The proposed amendment would increase the Reactor High Water Level Trip Level Setting for the Group 1 isolation. The change will allow an increase to the main steam isolation valve (MSIV) high water level isolation setpoint.

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

In accordance with 10 CFR 50.91, Boston Edison submits the following analysis addressing the no significant hazards consideration. The proposed changes do not:

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

Operation of the station in accordance with the proposed Trip Level Setting will not significantly increase the probability or consequences of an accident previously evaluated. The MSIV high water level isolation signal is provided to protect against rapid depressurization due to a pressure regulator malfunction during plant startup. The high water level isolation signal is not functional when the mode switch is in the RUN position. A high water level in the reactor vessel indicates that fuel is covered. Increasing the Trip Level Setting will have minimal effect on moisture carryover in the event of a pressure regulator failure at low reactor power. MSIV closure (Group 1) is initiated by low reactor pressure (810 psig) approximately 30 seconds into the event. The resulting reactor water level swell is not sufficient to reach the bottom elevation of the main steam lines.

The proposed Technical Specification allowable value for the Reactor Low Level Trip Level Setting and the Reactor Low Low Water Level Trip Level setting does not involve significant increase in the probability or consequence of an accident.

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

The proposed change does not affect the Group 1 isolation safety function. The change does not involve any plant hardware changes that could introduce any new failure modes or effects; thus, the change can not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed change does not affect the Group 1 isolation safety function. The proposed change is consistent with the FSAR [Final Safety Analysis Report] and Technical Specification basis associated with reactor vessel inventory control and main steam line flooding.

The proposed change to the instrument calibration range does not affect the margin of safety for systems or components affected by the change. Operating Pilgrim in accordance with the proposed Trip Level Setting does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

Attorney for licensee: W. S. Stowe, Esquire, Boston Edison Company, 800 Boylston Street, 36th Floor, Boston, Massachusetts 02199.

NRC Project Director: Walter R. Butler

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 amendment request: February 6, 1995

Description of amendment request: The change proposes to relocate the cycle specific core operating limits of Figure 3.1-1, Shutdown Margin Versus Boron Concentration, from Technical Specification (TS) 3.1.1.2, Shutdown Margins - Modes 3, 4, and 5, to the Core Operating Limits Report (COLR).

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

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

The proposed change of relocating TS Figure 3.1-1, Shutdown Margin Versus Boron Concentration to the COLR has no influence or impact to the probability or consequences of an accident. The revised TS will continue to implement the shutdown margin limits through reference to the Shutdown Margin Curve in the COLR. In addition, the COLR is subject to the existing controls of TS 6.9.1.6. Given that this change is an administrative relocation of the Shutdown Margin Curve to another TS controlled document, there would be no increase in the probability or consequences of an accident previously evaluated.

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

No safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. The TS will continue to require operation within the required core operating limits. 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 amendment does not involve a significant reduction in the margin of safety.

Relocation of the Shutdown Margin Curve to the TS controlled COLR has no effect on the core operating limits currently in force in TS 3.1.1.2. Future revisions to the Shutdown Margin Curve are governed by TS 6.9.1.6 which stipulates the specific TS that reference the COLR limits and the methodologies utilized in developing those limits. Given that the change is an administrative relocation of the Shutdown Margin Curve to another TS controlled document, 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: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605

Attorney for licensee: R. E. Jones, General Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: William H. Bateman

Duke Power Company, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: January 12, 1995

Description of amendment request: The proposed amendments would revise and clarify portions of Technical Specification (TS) Section 6.0, ''Administrative Controls,'' for the McGuire, Catawba, and Oconee nuclear stations. The licensee submitted a combined amendment request covering the three Duke Power nuclear stations. The proposed changes are described below.

1. Remove the specific assignment of responsibilities for the review, distribution, and approval activities contained in the Technical Review and Control Section of each station's TS. The proposed specifications state that these activities will be performed by a knowledgeable individual/organization. Approval of the affected documents is to be at the appropriate manager/superintendent level as specified in Duke administrative controls.

2. Move the requirement for the review of proposed changes in the stations' TS and Operating Licenses by the Duke Nuclear Safety Review Board (NSRB) to Duke administrative procedures (Selected Licensee Commitments documents) and change the wording of the requirements covering NSRB meeting frequency. The Oconee TS covering the NSRB are being rewritten to be consistent with McGuire and Catawba.

3. Add Technical Review and Control Program implementation and Plant Operations Review Committee (PORC) implementation to the list of required procedures and programs for each nuclear station.

4. Change or clarify certain TS administrative requirements covering technical review and control activities or records retention requirements.

5. For Oconee only, under ''Station Operating Procedures,'' revise the TS requirements covering the review and approval of station procedures and temporary procedure changes such that these are now consistent with the corresponding requirements for McGuire and Catawba.

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:

(It should be noted that the licensee submitted a combined analysis that covers McGuire, Catawba, and Oconee nuclear stations.)

Standard ι1. The proposed amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The provisions of these proposed amendments concern administrative changes in the stations' Technical Specifications involving the Technical Review and Control, Procedures and Programs/Station Operating Procedures, and Records Retention/Station Operating Records portions of the Administrative Controls Section. The requested changes primarily affect review and control activities, but also include other administrative changes affecting the approval of station procedures (Oconee only), records retention, and definition of the term ODCM [offsite dose calculation manual] (McGuire and [Catawba]). The provisions of the proposed amendment primarily involve the relocation of existing Technical Specifications review, distribution, or approval requirements to internal Duke administrative controls. However, implementation of the proposed amendment does involve changes to several review/distribution activities. Theses review/distribution activities are primarily for: 1) Proposed changes to the stations' Technical Specifications, 2) Proposed tests and experiments which affect nuclear safety and are not addressed in the stations' FSAR [Final Safety Analysis Report] or Technical Specifications, 3) Environmental radiological procedures, 4) Reportable events documentation and reports of violations of Technical Specifications, 5) Reports of special reviews and investigations, and 6) Reports of unplanned onsite releases of radiological material to the environs. Planned implementation of the proposed Technical Specifications amendments utilizing Selected Licensee Commitments will result in the above items being reviewed/received by a different organizational unit in the future. The organizational unit is to be either the recently initiated Plant Operations Review Committee (PORC) or the General Manager, Environmental Services. Personnel serving on the PORC, and the General Manager, Environmental Services will be qualified based upon education and experience to review the operational and technical considerations involved with the applicable items listed above. No required reviews are being eliminated by the requested amendments, only the organizational units responsible for performing the reviews will be changed. Future reviews of theses items under the auspices of the PORC or the General Manager, Environmental Services will maintain a quality level equivalent to that being currently achieved by Duke's Qualified Reviewer Program, the Station Managers, or the

Duke Nuclear Safety Review Board as applicable. Consequently, merely changing the organizational units performing future reviews, or making the additional administrative changes described above, results in no increase in the probability or consequences of an accident previously evaluated because the review function will continue to be conducted in an equivalent manner.

The implementing SLC will also permit proposed amendments to the stations' Technical Specifications and Operating Licenses to be approved for the Station Manager by a designee. However, this individual will occupy a position equivalent to, or higher, in the Duke organization as the Station manager.

Additionally, the proposed changes do not directly impact the design or operation of any plant systems or components any more so than the review and approval processes currently being conducted in accordance with existing approved Technical Specifications.

Standard ı2. The proposed amendments will not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes are administrative in nature and primarily cover the review, distribution, and/or approval function performed for items identified in existing Technical Specifications. The quality level of the future reviews will not decrease and the ability of Duke to identify the possibility for the concurrence of new or different kinds of accidents prior to implementation will be maintained. Of specific interest in the consideration of Standard ı2 is the review of proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications. The Technical Specifications required reviews of these tests and experiments are not being proposed for removal by these requested amendments. Only the organizational unit conducting the review of proposed tests and experiments is being changed by the requested amendments. The PORC, instead of the Station Manager, is being assigned the responsibility for conducting the reviews of proposed tests and experiments in the future. It is believed that the combined expertise of the PORC membership will enhance Duke's ability to identify potential situations which could possibly involve a new, or different, kind of accident.

Standard ı3. The proposed amendments will not involve a significant reduction in any margin of safety.

The changes contained in the requested amendments are administrative in nature and do not impact the design capabilities or operation of any plant structures, systems, or components. There will be no reduction in margin of safety as a result of implementing these requested amendments. Impact upon margin of safety is a consideration primarily included in the 10 CFR 50.59 evaluation process conducted for station procedures, procedure changes, and nuclear station modifications. The 10 CFR 50.59 evaluation process in conducted under the auspices of the Duke Qualified Reviewer Program and is not affected by these requested amendments. The impact on margin of safety for future Technical Specifications and Operating License changes will be reviewed by the PORC, but these reviews will be equivalent in quality to the reviews presently conducted by the Qualified Reviewers.

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: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242

NRC Project Director: Herbert N. Berkow

Duke Power Company, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: January 13, 1995

Description of amendment request: The proposed amendments would increase the surveillance test intervals and allowed outage times for Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) equipment based upon analyses by Westinghouse for the Westinghouse Owners Group and approved by the NRC. The proposed changes to the RTS and ESFAS instrumentation are based upon WCAP-10271, its supplements, and the NRC's safety evaluation reports.

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

Criterion 1 - Operation of McGuire in accordance with the proposed license amendment[s] [do] not involve a significant increase in the probability or consequences of an accident previously evaluated.

The determination that the results of the proposed changes are within all acceptable criteria was established in the SERs prepared for WCAP-10271, WCAP-10271 Supplement 1, WCAP-10271 Supplement 2, and WCAP-10271 Supplement 2, Revision 1 issued by letters dated February 21, 1985, February 22, 1989, and April 30, 1990. Implementation of the proposed changes is expected to result in an acceptable increase in total RTS yearly unavailability. This increase, which is primarily due to less frequent surveillance, results in an increase of similar magnitude in the probability of an Anticipated Transient Without Scram (ATWS) and in the probability of core melt resulting from an ATWS and also results in a small increase in core damage frequency (CDF) due to ESFAS unavailability.

Implementation of the proposed changes is expected to result in a significant reduction in the probability of core melt from inadvertent reactor trips. This is a result of a reduction in the number of inadvertent reactor trips (0.5 fewer inadvertent reactor trips per unit per year) occurring during testing of RTS instrumentation. This reduction is primarily attributable to testing in bypass and less frequent surveillance.

The reduction in core melt frequency from inadvertent reactor trips is sufficiently large to counter the increase in ATWS core melt probability resulting in an overall reduction in total core melt probability.

The values determined by the WOG and presented in the WCAP for the increase in CDF were verified by Brookhaven National Laboratory (BNL) as part of an audit and sensitivity analysis for the NRC staff. Based on the small value of the increase compared to the range of uncertainty in the CDF, the increase is considered acceptable.

Changes to surveillance test frequencies for the RTS [reactor trip system] interlocks do not represent a significant reduction in testing. The currently specified test interval for interlock channels allows the surveillance requirement to be satisfied by verifying that the permissive logic is in its required state using the permissive annunciator window. The surveillance as currently required only verifies the status of the permissive logic and does not address verification of channel setpoint or operability. The setpoint verification and channel operability are verified after a refueling shutdown. The definition of the channel check includes comparison of the channel status with other channels for the same parameter. The requirement to routinely verify permissive status is a different consideration than the availability of trip or actuation channels which are required to change state on the occurrence of an event and for which the function availability is more dependent on the surveillance interval. The change in surveillance requirement to at least once every refueling does not therefore represent a significant change in channel surveillance and does not involve a significant increase in unavailability of the RTS.The proposed changes do not result in an increase in the severity or consequences of an accident previously evaluated. Implementation of the proposed changes affects the probability of failure of the RTS but does not alter the manner in which protection is afforded nor the manner in which limiting criteria are established.

Criterion 2 - The proposed license amendment[s] [do] not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not result in a change in the manner in which the RTS provides plant protection. No change is being made which alters the functioning of the RTS (other than in a test mode). Rather, the likelihood or probability of the RTS functioning properly is affected as described above. Therefore, the proposed changes do not create the possibility of a new or different kind of accident.

The proposed changes do not involve hardware changes except those necessary to implement testing in bypass. Some existing instrumentation is designed to be tested in bypass and current Technical Specifications allow testing in bypass. Testing in bypass is also recognized by IEEE standards. Therefore, testing in bypass has been previously approved and implementation of the proposed changes for testing in bypass does not create the possibility of a new or different kind of accident from any previously evaluated. Furthermore, since the other proposed changes do not alter the functioning of the RTS, the possibility of a new or different kind of accident from any previously evaluated has not been created.

Criterion 3 - The proposed license amendment[s] [do] not involve a significant reduction in a margin of safety.

The proposed changes do not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The impact of reduced testing other than as addressed above is to allow a longer time interval over which instrument uncertainties (e.g., drift) may act. Experience has shown that the initial uncertainty assumptions are valid for reduced testing.

Implementation of the proposed changes is expected to result in an overall improvement in safety by:

1) Less frequent testing will result in fewer inadvertent reactor trips and actuation of Engineered Safety Features Actuation System components.

2) Higher quality repairs leading to improved equipment reliability due to longer allowable repair times.

3) Improvements in the effectiveness of the operating staff in monitoring and controlling plant operation. This is due to less frequent distraction of the operator and shift supervisor to attend to instrumentation testing.

The foregoing analysis demonstrates that the proposed amendment[s] to McGuire's Technical Specifications [do] not involve a significant increase in the probability or consequences of a previously evaluated accident, [do] not create the possibility of a new or different kind of accident, and [do] not involve a significant reduction in a margin of safety.

Based upon the preceding analysis, Duke Power Company concludes that the proposed amendment[s] [do] not involve a significant hazards consideration.

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

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: January 12, 1995

Description of amendment request: The proposed amendments would revise and clarify portions of Technical Specification (TS) Section 6.0, ''Administrative Controls,'' for the McGuire, Catawba, and Oconee nuclear stations. The licensee submitted a combined amendment request covering the three Duke Power nuclear stations. The proposed changes are described below.

1. Remove the specific assignment of responsibilities for the review, distribution, and approval activities contained in the Technical Review and Control Section of each station's TS. The proposed specifications state that these activities will be performed by a knowledgeable individual/organization. Approval of the affected documents is to be at the appropriate manager/superintendent level as specified in Duke administrative controls.

2. Move the requirement for the review of proposed changes in the stations' TS and Operating Licenses by the Duke Nuclear Safety Review Board (NSRB) to Duke administrative procedures (Selected Licensee Commitments documents) and change the wording of the requirements covering NSRB meeting frequency. The Oconee TS covering the NSRB are being rewritten to be consistent with McGuire and Catawba.

3. Add Technical Review and Control Program implementation and Plant Operations Review Committee (PORC) implementation to the list of required procedures and programs for each nuclear station.

4. Change or clarify certain TS administrative requirements covering technical review and control activities or records retention requirements.

5. For Oconee only, under ''Station Operating Procedures,'' revise the TS requirements covering the review and approval of station procedures and temporary procedure changes such that these are now consistent with the corresponding requirements for McGuire and Catawba.

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:

(It should be noted that the licensee submitted a combined analysis that covers McGuire, Catawba, and Oconee nuclear stations.)

Standard ι1. The proposed amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The provisions of these proposed amendments concern administrative changes in the stations' Technical Specifications involving the Technical Review and Control, Procedures and Programs/Station Operating Procedures, and Records Retention/Station Operating Records portions of the Administrative Controls Section. The requested changes primarily affect review and control activities, but also include other administrative changes affecting the approval of station procedures (Oconee only), records retention, and definition of the term ODCM [offsite dose calculation manual] (McGuire and [Catawba]). The provisions of the proposed amendment primarily involve the relocation of existing Technical Specifications review, distribution, or approval requirements to internal Duke administrative controls. However, implementation of the proposed amendment does involve changes to several review/distribution activities. These review/distribution activities are primarily for: 1) Proposed changes to the stations' Technical Specifications, 2) Proposed tests and experiments which affect nuclear safety and are not addressed in the stations' FSAR [Final Safety Analysis Report] or Technical Specifications, 3) Environmental radiological procedures, 4) Reportable events documentation and reports of violations of Technical Specifications, 5) Reports of special reviews and investigations, and 6) Reports of unplanned onsite releases of radiological material to the environs. Planned implementation of the proposed Technical Specifications amendments utilizing Selected Licensee Commitments will result in the above items being reviewed/received by a different organizational unit in the future. The organizational unit is to be either the recently initiated Plant Operations Review Committee (PORC) or the General Manager, Environmental Services. Personnel serving on the PORC, and the General Manager, Environmental Services will be qualified based upon education and experience to review the operational and technical considerations involved with the applicable items listed above. No required reviews are being eliminated by the requested amendments, only the organizational units responsible for performing the reviews will be changed. Future reviews of these items under the auspices of the PORC or the General Manager, Environmental Services will maintain a quality level equivalent to that being currently achieved by Duke's Qualified Reviewer Program, the Station Managers, or the Duke Nuclear Safety Review Board as applicable. Consequently, merely changing the organizational units performing future reviews, or making the additional administrative changes described above, results in no increase in the probability or consequences of an accident previously evaluated because the review function will continue to be conducted in an equivalent manner.

The implementing SLC will also permit proposed amendments to the stations' Technical Specifications and Operating Licenses to be approved for the Station Manager by a designee. However, this individual will occupy a position equivalent to, or higher, in the Duke organization as the Station Manager.

Additionally, the proposed changes do not directly impact the design or operation of any plant systems or components any more so than the review and approval processes currently being conducted in accordance with existing approved Technical Specifications.

Standard ι2. The proposed amendments will not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes are administrative in nature and primarily cover the review, distribution, and/or approval function performed for items identified in existing Technical Specifications. The quality level of the future reviews will not decrease and the ability of Duke to identify the possibility for the occurrence of new or different kinds of accidents prior to implementation will be maintained. Of specific interest in the consideration of Standard ι2 is the review of proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications. The Technical Specifications required reviews of these tests and experiments are not being proposed for removal by these requested amendments. Only the organizational unit conducting the review of proposed tests and experiments is being changed by the requested amendments. The PORC, instead of the Station Manager, is being assigned the responsibility for conducting the reviews of proposed tests and experiments in the future. It is believed that the combined expertise of the PORC membership will enhance Duke's ability to identify potential situations which could possibly involve a new, or different, kind of accident.

Standard ι3. The proposed amendments will not involve a significant reduction in any margin of safety.

The changes contained in the requested amendments are administrative in nature and do not impact the design capabilities or operation of any plant structures, systems, or components. There will be no reduction in margin of safety as a result of implementing these requested amendments. Impact upon margin of safety is a consideration primarily included in the 10 CFR 50.59 evaluation process conducted for station procedures, procedure changes, and nuclear station modifications. The 10 CFR 50.59 evaluation process is conducted under the auspices of the Duke Qualified Reviewer Program and is not affected by these requested amendments. The impact on margin of safety for future Technical Specifications and Operating License changes will be reviewed by the PORC, but these reviews will be equivalent in quality to the reviews presently conducted by the Qualified Reviewers.

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

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691

Attorney for licensee: J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20036

NRC Project Director: Herbert N. Berkow

Entergy Operations Inc., Docket No. 50-382, Waterford Steam ElectricStation, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: January 27, 1995

Description of amendment request: The requested change would modify Section 5.3.1, Fuel Assemblies, of the Waterford 3 technical specifications. The requested change increases the maximum enrichment for the spent fuel pool and containment temporary storage rack from 4.1 to 4.9 weight percent U-235 when fuel assemblies contain fixed poisons. Waterford 3 plans to use higher enriched fuel in the next fuel cycle (Cycle 8) to meet the energy plans and maintain a reload batch size similar to that used in Cycles 6 and 7.

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

The proposed change will increase the fuel enrichment limit in order to meetthe cycle energy requirements while maintaining fuel batch sizes consistent with previous cycle designs. The calculated k-effective, including uncertainties, demonstrate substantial margin to criticality in the storage racks for both normal and accident conditions. No changes to the facility are required. No new modes of operating the fuel storage or transfer systems are required, except a restriction to limit the use of the new fuel vault to fuel with a maximum enrichment of 4.1 weight percent U-235. This restriction will be implemented by administrative controls. Since the plant equipment and operation are essentially the same, there is no significant increase in the probability of a criticality accident. Since a criticality event is demonstrated to be unfeasible, there are no increased adverse consequences for such a postulated event.

As previously discussed, the proposed change will not result in a physical change to the facility nor will it result in a significant change to the operation of the facility; therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change has been analyzed to establish a k-effective, including uncertainties, at or below the NRC criticality acceptance criteria of k-effective below 0.95 including uncertainties at the 95/95 probability/confidence level; therefore, there is no reduction in the margin of safety.

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

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, Louisiana 70122

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street N.W., Washington, D.C. 20005-3502

NRC Project Director: William D. Beckner

GPU Nuclear Corporation, et al., Docket No. 50-289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

Date of amendment request: January 16, 1995

Description of amendment request: The proposed amendment would revise the TMI-1 Technical Specifications (TS) to incorporate certain improvements from the Revised Standard Technical Specifications (TS) for Babcock & Wilcox nuclear power plants (NUREG-1430). The amendment would also change the bases incorporating the results of analyses to support allowance for drift of the pressurizer code safety valve setpoint. One of the proposed STS improvements involves a change to Chapter 6, Administrative Controls, affecting both TMI-1 and TMI-2 TSs. A separate notice of consideration of issuance of amendment to facility operating license is being issued for the proposed TMI-2 TSs Change.

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

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

The proposed amendments involve a) an administrative change to both the TMI-1 and TMI-2 Technical Specifications which is consistent with the B&W Standard Technical Specifications (STS), NUREG-1430, and b) changes to the TMI-1 Technical Specifications which are consistent with the STS. This change does not involve any change to system or equipment configuration. The proposed amendment revises certain surveillance requirements, extends certain surveillance intervals as evaluated above, or involves changes that are purely administrative. The reliability of systems and components relied upon to prevent or mitigate the consequences of accidents previously evaluated is not degraded by the proposed changes. Assurance of system and equipment availability is maintained. Therefore, this change does not increase the probability of occurrence or the consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes only involve changes to surveillance requirements that are consistent with STS and with the ASME Code. No new failure modes are created and thus the changes are bounded by accidents previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not

involve a significant reduction in a margin of safety. Each of these changes is compatible with the STS and has been evaluated to preserve the level of safety assured by the current TS.

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: Government Publications Section, State Library of Pennsylvania, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

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

NRC Project Director: Phillip F. McKee

Gulf States Utilities Company, Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: January 20, 1995

Description of amendment request: The proposed amendment would revise the fire hazards analysis for the River Bend Station (RBS) by allowing a deviation from 10 CFR 50, Appendix R, Section III.G.3 with respect to the requirement for a fixed fire suppression system in fire area C-17. This area houses the control building heating, ventilation and air conditioning (HVAC) systems and the loss due to a fire could cause the loss of main control room habitability. C-17 does not have a fixed fire suppression system but depends upon the use of the existing remote shutdown system as described in the updated safety analysis report (USAR).

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

The event of concern is a fire in fire area C-17. The low fire loading and sparse concentration of exposed combustible material in fire area C-17 would limit fire spread. However, for this scenario all equipment in fire area C-17 will be assumed lost. Fire area C-17 contains the air handling units for the main control room envelope. The loss of both air handling units would cause the control building chillers to stop running due to a logic tie requiring air flow through the air handling equipment for the chilled water system to operate during normal operation. The loss of the HVAC system in the control building would cause the main control room and the equipment rooms to begin heating up if exposed to design summer conditions. Operator actions can be accomplished to minimize the heat up rates for the rooms prior to the areas reaching equipment temperature limits. This would allow the operators to begin the shutdown process from the main control room. If the main control room continued to heat up, the operators could accomplish the shutdown using the remote shutdown system. HVAC for the remote shutdown panel is located in fire area C-4 and would not be damaged by a fire in fire area C-17. Operation of the control building HVAC system from the remote shutdown panel bypasses the logic between the chilled water system and the air handling system. This would allow restart of the HVAC system for all areas except the main control room. The scenario would conclude in a manner similar to that described in RBS USAR Appendix 15A, Event 52, ''Reactor Shutdown From Outside Main Control Room.''

In summary, the probability of a fire occurring in fire area C-17 is not increased. However, if a fire were to occur in fire area C-17 which caused the loss of main control room HVAC, the remote shutdown system would provide an acceptable method of shutdown. The low fire loading and sparse concentration of exposed combustible material in fire area C-17 would limit fire spread. Therefore, this request does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The event of concern is a fire in fire area C-17. Fire area C-17 does not have a fixed suppression system as required by 10 CFR 50, Appendix R, Section III.G.3. Fire suppression systems are generally used to limit fire spread, once the heat of the fire opens thermally sensitive sprinklers. The low fire loading and sparse concentration of exposed combustible material in fire area C-17 would limit fire spread. However, for the purpose of event analysis, all equipment in fire area C-17 is assumed lost. Thus a fire in fire area C-17 is bounded by the same analysis with or

without a fixed suppression system in terms of equipment availability.

The proposed method of shutdown for a fire in fire area C-17 will be changed in that the remote shutdown system will be credited. Use of the remote shutdown system is bounded by RBS USAR Appendix 15A, Event 52, ''Reactor Shutdown From Outside Main Control Room.'' The HVAC for the remote shutdown panel is located in fire area C-4 and would be undamaged by a fire in fire area C-17. Operation of the control building HVAC system from the remote shutdown panel bypasses the logic between the chilled water system and the air handling system. This would allow restart of the HVAC system for all areas except the main control room.

In summary, if a fire were to occur in fire area C-17 which caused the loss of main control room HVAC, the remote shutdown system would provide an acceptable method of shutdown. Since, for the purpose of event analysis, all equipment in fire area C-17 is assumed lost, a fire in fire area C-17 is bounded by the same analysis with or without a fixed suppression system in terms of equipment availability. Therefore, this request does not create the possibility of occurrence of a new or different kind of accident from any accident previously evaluated.

3) The request does not involve a significant reduction in a margin of safety.

In this case, the margin of safety is implicit rather than being explicitly expressed as a numerical value. An implicit margin of safety involves conditions for NRC acceptance. Since the RBS Technical Specification Bases do not specifically address a margin of safety for fire protection, the SAR, the NRC's Safety Evaluation Report (SER), and appropriate other licensing basis documents were reviewed to determine if the proposed change would result in a reduction in a margin of safety. As stated, in part, in Attachment 4 to NPF-47:

EOI shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility through Amendment 22 and as approved in the SER dated May 1984 and Supplement 3 dated August 1985 subject to provisions 2 and 3....

As discussed in the Reason for Request, SSER 3 dated August 1985 states, in part:

On the basis of its evaluation the staff finds that the applicant's fire protection program with approved deviations is in conformance with the guidelines of BTP CMEB 9.5-1, sections III.G, III.J, and III.O of Appendix R to 10CFR50, and GDC 3, and is, therefore, acceptable.

Thus, the margin of safety in this case can be defined as conformance with the specified fire protection guidelines. 10 CFR 50, Appendix R, Section III.G.3, requires, in part, that alternative shutdown capability be provided for areas where adequate separation of redundant safe shutdown components cannot be provided. In addition, fire detection and a fixed fire suppression system must be installed in the area, room, or zone under consideration. Since fire area C-17 does not have a fixed suppression system, use of the remote shutdown system for a fire in this fire area would deviate from the requirements of 10 CFR 50, Appendix R, Section III.G.3. However, as discussed previously, the low fire loading and sparse amount of exposed combustibles compensate for the lack of a fixed fire suppression system. There is no adverse impact on the ability to achieve and maintain safe shutdown. Therefore, this request 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: Government Documents Department, Louisiana State University, Baton Rouge, Louisiana 70803

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005

NRC Project Director: William D. Beckner

Northeast Nuclear Energy Company (NNECO), Docket No. 50-245, Millstone Nuclear Power Station, Unit 1, New London County, Connecticut

Date of amendment request: July 28, 1994

Description of amendment request: The proposed amendment would add a footnote to Technical Specifcaiton 3.5.C. The footnote would state that the operability of the feedwater coolant injection (FWCI) system be independent of its seismic capability.

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 change in accordance with 10CFR50.92 and concluded that the change does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed change does not involve an SHC because the change would not:

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

Any postulated failure in the non-seismic portion of the FWCI subsystem may result in a loss of feedwater flow transient. However comparing the probability of occurrence of a seismic event, any increase in the probability of occurrence of a loss of feedwater event would be small. The proposed change would have no impact on the probability of occurrence of any other accident, including LOCAs [loss of coolant accidents].

The FWCI subsystem will continue to be maintained as QA Category 1 (except for the seismic attribute). Therefore, it will remain available for accident mitigation for most scenarios. Nevertheless, LOCA analyses have been reevaluated to demonstrate that FWCI is not necessary to show compliance with 10CFR50.46. Potentially limiting LOCA scenarios have been analyzed without the FWCI subsystem using an approved LOCA methodology. An active single failure was postulated in addition to not taking credit for the FWCI subsystem. Based on the results of these analyses, the current design basis large and small break LOCAs remain bounding. Moreover, FWCI is not credited in mitigating any of the non-LOCA transients/accidents.

Safe shutdown following a seismic event can be achieved using the LPCI [low pressure coolant injection] and ESW [emergency service water] systems, and the SRVs [safety relief valves], which are all seismically qualified. Therefore, the FWCI system is not required to mitigate a seismic event.

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

Seismic reclassification of portions of FWCI does not create the possibility of a new kind of an accident. The portion of the piping up to the second isolation valve (from the RPV [reactor pressure vessel]), is seismically qualified and will remain classified as seismic. This ensures that a postulated failure in the non-seismic portion of piping or components does not degrade containment integrity or result in a blowdown of the RPV. Consequential and environmental effects of a FW [feedwater] piping failure have been analyzed in the HELB [high energy line break] program and have been found to be acceptable.

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

All accidents, including LOCAs, can be mitigated without using FWCI. FWCI is also not necessary for safe shutdown following a seismic event. The intended function of the FWCI subsystem is to reduce the likelihood of core uncovery during the lifetime of the plant. The CS [core spray] and LPCI subsystems provide redundant and diverse means of injecting water to the RPV. The FWCI subsystem provides an additional diverse means to inject water. Since FWCI will be maintained QA Category 1 (except for the seismic attribute), it will continue to provide the additional diversity to the injection systems. Considering the intended function of the subsystem and the credit taken in the accident analysis, reclassifying FWCI to be non-seismic does not significantly reduce the margin of safety.

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

Local Public Document Room location: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141-0270.

NRC Project Director: Phillip F. McKee

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 amendment requests: January 9, 1995, as supplemented February 7, 1995

Description of amendment requests: The proposed amendments would revise Prairie Island Nuclear Generating Plant Technical Specification (TS) 4.12, ‘‘Steam Generator Tube Surveillance,’’ to incorporate revised acceptance criteria for steam generator tubes with degradation in the tubesheet roll expansion region. These criteria for steam generator tube acceptance were developed by Westinghouse Electric Corporation and are known as F* (≥F-Star’’) and L* (≥L-Star’’). These criteria would be utilized to avoid unnecessary plugging and sleeving of steam generator tubes.

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

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

F Steam Generator Tube Repair Criteria*

The supporting technical and safety evaluations of the subject criterion demonstrate that the presence of the tubesheet will enhance the tube integrity in the region of the hardroll by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and tube collapse is strengthened by the presence of the tubesheet in that region. The results of hardrolling of the tube into the tubesheet is an interference fit between the tube and the tubesheet. Tube rupture cannot occur because the contact between the tube and tubesheet does not permit sufficient movement of tube material. The radial preload developed by the rolling process will secure a postulated separated tube end within the tubesheet during all plant conditions. In a similar manner, the tubesheet does not permit sufficient movement of tube material to permit buckling collapse of the tube during postulated LOCA [loss-of-coolant accident] loadings.

The F* length of roll expansion is sufficient to preclude tube pullout from tube degradation located below the F* distance, regardless of the extent of the tube degradation. The existing Technical Specification leakage rate requirements and accident analysis assumptions remain unchanged in the unlikely event that significant leakage from this region does occur. As noted above, tube rupture and pullout is not expected for tubes using the F* criterion. Any leakage out of the tube from within the tubesheet at any elevation in the tubesheet is fully bounded by the existing steam generator tube rupture analysis included in the Prairie Island Plant USAR [Updated Safety Analysis Report]. For plants with partial depth roll expansion like Prairie Island, a postulated tube separation within the tube near the top of the roll expansion (with subsequent limited tube axial displacement) would not be expected to result in coolant release rates equal to those assumed in the USAR for a steam generator tube rupture event due to the limited gap between the tube and tubesheet. The proposed plugging criterion does not adversely impact any other previously evaluated design basis accident.

Leakage testing of roll expanded tubes indicates that for roll lengths approximately equal to the F* distance, any postulated faulted condition primary to secondary leakage from F* tubes would be insignificant.

L Steam Generator Tube Repair Criteria*

The presence of the tubesheet enhances steam generator tube integrity in the region of the hardroll by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and tube collapse is strengthened by the presence of the tubesheet in that region. The result of the hardroll of the tube into the tubesheet is an interference fit between the tube and the tubesheet. Tube rupture cannot occur because the contact between the tube and tubesheet does not permit sufficient movement of tube materials. In a similar manner, the tubesheet does not permit sufficient movement of tube material to permit buckling collapse of the tube during postulated LOCA loadings.

The type of degradation for which the L* criteria has been developed (cracking with an axial or near axial orientation) has been found not to significantly reduce the axial strength of a tube. An evaluation including analysis and testing has been done to determine the strength reduction for the axial loads with simulated axial and near axial cracks. This evaluation provided the basis for the acceptance criteria for tube degradation subject to the L* criteria.

The length of roll expansion above L* is sufficient to preclude significant leakage from tube degradation located below the L* distance. The existing Technical Specification leakage rate requirements and accident analysis assumptions remain unchanged in the unlikely event that significant leakage from this region does occur. As noted above, tube rupture and pullout is not expected for tubes using the alternate plugging criteria.

Any leakage out of the tube from within the tubesheet at any elevation in the tubesheet is fully bounded by the existing steam generator tube rupture analysis included in the Prairie Island Updated Safety Analysis Report. The proposed alternate plugging criteria do not adversely impact any other previously evaluated design basis accident.

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

*F**

Implementation of the proposed F* criterion does not introduce any significant changes to the plant design basis. Use of the criterion does not provide a mechanism to initiate an accident outside of the region of the expanded portion of the tube. Any hypothetical accident as a result of any tube degradation in the expanded portion of the tube would be bounded by the existing tube rupture accident analysis. Tube bundle structural integrity will be maintained. Tube bundle leaktightness will be maintained such that any postulated accident leakage from F* tubes will be negligible with regards to offsite doses.

*L**

Implementation of the proposed alternate tubesheet tube plugging criteria does not introduce changes to the plant design basis. Use of the criteria does not provide a mechanism to result in an accident outside of the region of the tubesheet expansion. Any hypothetical accident as a result of any tube degradation in the expanded portion of the tube would be bounded by the existing tube rupture accident analysis.

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

*F**

The use of the F* criterion has been demonstrated to maintain the integrity of the tube bundle commensurate with the requirements of Reg Guide 1.121 [≥Bases for Plugging Degraded PWR Steam Generator Tubes≥] (intended for indications in the free span of tubes) and the primary to secondary pressure boundary under normal and postulated accident conditions. Acceptable tube degradation for the F* criterion is any degradation indication in the tubesheet region, more than the F* distance below the bottom of the transition between the roll expansion and the unexpanded tube. The safety factors used in the verification of the strength of the degraded tube are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in steam generator design. The F* distance has been verified by testing to be greater than the length of roll expansion required to preclude both tube pullout and significant leakage during normal and postulated accident conditions. Resistance to tube pullout is based upon the primary to secondary pressure differential as it acts on the surface area of the tube, which includes the tube wall cross-section, in addition to the inner diameter based area of the tube. The leak testing acceptance criteria are based on the primary to secondary leakage limit in the Technical Specifications and the leakage assumptions used in the USAR accident analysis.

Implementation of the tubesheet plugging criterion will decrease the number of tubes which must be taken out of service with tube plugs or repaired with sleeves. Both plugs and sleeves reduce the RCS (reactor coolant system) flow margin; thus, implementation of the F* criterion will maintain the margin of flow that would otherwise be reduced in the event of increased plugging or sleeving.

Based on the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety as defined in the USAR or the Technical Specification Bases.

*L**

The use of the alternate tubesheet plugging criteria has been demonstrated to maintain the integrity of the tube bundle commensurate with the requirements of Reg. Guide 1.121 for indications in the free span of tubes and the primary to secondary pressure boundary under normal and postulated accident conditions. Acceptable tube degradation for the L* criteria is any degradation indication with axial or nearly axial cracking in the tubesheet region, more than the L* distance below the bottom of the transition between the roll expansion and the unexpended tube. For tubes with axial or nearly axial cracks the strength of the tube relative to an axial load would not be reduced below the strength required to resist potential axial loads. The safety factors used in the verification of the strength of the degraded tube are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in steam generator design. The L* distance has been verified by testing to be greater than the length of roll expansion required to preclude significant leakage during normal and postulated accident conditions. The leak testing acceptance criteria are based on the primary to secondary leakage limit in the Technical Specifications and the leakage assumptions used in the USAR accident analyses.

Implementation of the proposed tubesheet plugging criteria will decrease the number of tubes which must be taken out of service with tube plugs or repaired with sleeves. Both plugs and sleeves reduce the RCS flow margin, thus implementation of the alternate plugging criteria will maintain the margin of flow that would otherwise be reduced in the event of increased plugging or sleeving.

Based on the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety as defined in the Updated Safety Analysis Report or the bases of the Technical Specifications.

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 requests involve no significant hazards consideration. This notice supersedes the staff's previous notice which was published in the **Federal Register** February 1, 1995 (60 FR 6307).

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

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

NRC Project Director: Cynthia Carpenter, Acting

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 amendment requests: February 23, 1995

Description of amendment requests: The proposed amendments would revise the wording in the Prairie Island technical specifications to allow implementation of exemptions to the schedule requirements of 10 CFR Part 50, Appendix J. A related exemption request would grant temporary relief from the requirements of 10 CFR Part 50, Appendix J, Section III.D.1.(a) which requires Prairie Island Unit 2 to perform a Type A test in the May 1995 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 amendment[s] will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment is an administrative change which allows implementation of approved exemptions to the regulations and by itself does not change any retest schedules.

Therefore, the probability or consequences of an accident previously evaluated are not affected by the proposed amendment.

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

The proposed amendment is an administrative change which allows implementation of approved exemptions to the regulations and by itself does not change any retest schedules.

Therefore, the possibility of a new or different kind of accident from any accident previously evaluated would not be created by the proposed amendment.

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

The proposed amendment is an administrative change which allows implementation of approved exemptions to the regulations and by itself does not change any retest schedules.

Therefore, a significant reduction in the margin of safety would not be involved with the proposed amendment.

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 requests involve no significant hazards consideration.

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

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

NRC Project Director: Cynthia Carpenter, Acting

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

Date of amendment request: February 10, 1995

Description of amendment request: The proposed amendment to the technical specifications (TSs) would relocate the requirements for the incore instrumentation (ICI) system from the TS to the Updated Safety Analysis Report (USAR).

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

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

The Incore Instrumentation (ICI) System is used to measure core power distribution for the purpose of Limiting Conditions for Operation (LCO) monitoring of Technical Specification (TS) limits on linear heat rate, unrodded planer radial peaking factor, unrodded integrated radial peaking factor, and azimuthal power tilt. The ICI System has no safety purpose itself; it measures parameters which have safety significance. No change to the monitored parameters is proposed. The proposed changes will relocate requirements on the number and distribution of incore detectors used by the ICI System when measuring these parameters from the TS to the Updated Safety Analysis Report (USAR). Changes to the requirements can be made without NRC approval when the changes meet the criteria of 10 CFR 50.59. Changes to the ICI System requirements that do not meet the criteria of 10 CFR 50.59 must be approved by the NRC by license amendment.

Relocation of the requirements on the ICI System from the TS to the USAR does not increase the probability or consequences of any accident previously analyzed because the ICI System is neither a precursor nor a mitigator for any analyzed accident. The ICI System is used to ensure that operation within the LCOs for linear heat rate, unrodded planer radial peaking factor, unrodded integrated radial peaking factor, and azimuthal power tilt is maintained. However, its operation serves no mitigation function associated with any USAR Section 14 accident analysis. The parameters measured by the ICI System are important parameters in many accident analyses; however, this proposed change does not remove or revise the limits on these parameters.

Additionally, it is proposed to revise TS 2.10.4(1)(b) to clarify its requirements. Currently TS 2.10.4(1) part (b) applies while operating under the provisions of part (a) if the plant computer incore detector alarms become inoperable. This is incorrect in that part (a) applies when the linear heat rate is being monitored by the ICI System and the linear heat rate is exceeding its limits as indicated by valid detector alarms. Part (b) of this specification should apply only if the linear heat rate is being monitored by the ICI System, is within its limits, and the plant computer incore detector alarms are inoperable.

Administrative changes are also proposed which correct grammar and renumber/relocate portions of the TS and bases to other TS, to correspond to the proposed change to relocate ICI System requirements.

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

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

The ICI System will continue to be used to monitor TS limits on core power distribution. There will be no physical alterations to the plant configuration, changes to setpoint values, or changes to the implementation of setpoints or limits as a result of this proposed change.

The proposed change to TS 2.10.4(1)(b) only clarifies its requirements. The proposed change is more restrictive in that TS 2.10.4(1)(b), as currently written, could be interpreted to allow continued operation for up to seven days with the linear heat rate exceeding its limits. The proposed change clarifies this specification to ensure that TS 2.10.4(1)(a) is applied if the linear heat rate is exceeded while being monitored by the ICI System. TS 2.10.4(1)(a) requires that the linear heat rate be restored within one hour or a plant shutdown initiated.

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

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

The ICI System is used to measure core power distribution parameters which are a direct measure of the margin of safety. The limits on these parameters are not changed. Therefore, the proposed change (i.e., relocation of the ICI System operability requirements to the USAR and/or plant procedures) does not involve a significant reduction in a margin of safety.

The proposed change to TS 2.10.4(1)(b) helps ensure that the margin of safety is maintained by clarifying when the TS is applicable. This clarification ensures that the more restrictive actions of TS 2.10.4(1)(a) are taken if the linear heat rate is exceeded while being monitored by the ICI System. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: LeBoeuf, Lamb, Leiby, and MacRae, 1875 Connecticut Avenue, NW., Washington, DC 20009-5728

NRC Project Director: Theodore R. Quay

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

Date of amendment requests: December 30, 1994 (Reference LAR 94-12)

Description of amendment requests: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, to revise TS 2.2, 3/4.3.1, 3/4.3.2, 3/4.3.3, 3/4.4.4, 3/4.4.9, 3/4.5.2, 3/4.8.1, 3/4.8.2, 3/4.9.2, 3/4.9.9, and 3/4.10.3. The specific TS changes proposed are as follows:

(1) The TS issued in License Amendments (LAs) 84/83 would be changed to (a) revise the value of the overpower Delta-temperature (OPDT) constant K6 in TS 2.2.1, Table 2.2-1, Note 3; (b) revise the reactor coolant system (RCS) loop Delta-T function; and (c) make editorial corrections for clarification and consistency to TS 2.2.1 (and TS 2.2.1 Bases), TS 3/4.3.1, and TS 3/4.3.2.

In revising the RCS loop Delta-T function, the licensee would (a) incorporate the 0.99 multiplying factor listed in TS 2.2.1, Table 2.2-1, Note 5, and TS 3/4.3.2, Table 3.3-4, Note 2, into constants B1 through B4; (b) change ''Steam Generator (SG) Water Level Low-Low'' in TS 3/4.3.2, Table 3.3-3 and Table 4.3-2, Functional Unit 6.c, ''Auxiliary Feedwater'' (AFW), by deleting the Mode 3 applicability of the RCS loop Delta-T function and by adding a footnote to the Mode 3 applicability of the SG water level low-low function requiring that the trip time delay (TTD) associated with the SG water level low-low channel be less than or equal to 464.1 seconds; (c) change TS 3/4.3.1, Table 3.3-1, Action 27, and TS 3/4.3.2, Table 3.3-3, Action 29, by allowing up to four RCS loop Delta-T channels to be inoperable with the TTD threshold power level for zero seconds time adjusted to 0-percent rated thermal power (RTP) and by allowing the affected SG water level low-low channels to be placed in the tripped condition, with one inoperable RCS loop Delta-T channel; and (d) change the Table 3.3-1 and Table 3.3-3 ''Channels to Trip'' and ''Minimum Channels Operable'' columns to not applicable (N.A.).

(2) The TS issued in LAs 70/69 would be changed to (a) delete references to the plant vent noble gas activity monitors (RM-14A and RM-14B) and footnote references to applicability of the containment ventilation exhaust radiation monitors (RM-44A and RM-44B) in TS Tables 3.3-3, 3.3-4, 3.3-5, 3.3-6, 4.3-2, and 4.3-3 and TS 4.9.9; and (b) revise the ''Trip Setpoint and Allowable Values'' column in TS Table 3.3-4, Functional Unit 3.c.4), to reference the offsite dose calculation procedure (ODCP).

(3) Cycle-specific information in TS 4.3.2.1, TS 3.3.3.6, TS 4.4.4.1, TS 4.5.2, TS 3.8.1.1, TS 3.8.2.1, and TS 3.8.2.2 that is no longer necessary would be deleted.

(4) The word ''analog'' would be deleted from TS 4.4.9.3.1, TS 4.9.2, and TS 4.10.3.2.

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

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

The proposed change to the OPDT constant K6 is conservative and will not cause any design or analysis acceptance criteria to be exceeded. There is no effect on the structural and functional integrity of any plant system. The OPDT function is part of the accident mitigation response and is not itself an initiator for any transient. This change does not affect the integrity of the fission product barriers for mitigation of radiological dose consequences as a result of an accident.

The proposed change to incorporate the 0.99 multiplier into the TTD constants is an administrative change and has no effect on plant operation. The proposed change to delete Mode 3 applicability of the RCS Loop Delta-T function does not affect any design or analysis results. Allowing up to 4 RCS Loop Delta-T channels to be inoperable with the TTD threshold power level for zero seconds time delay adjusted to 0% RTP is conservative with respect to ESFs [engineered safety features] and reactor trip actuation time. Allowing the SG [steam generator] water level low-low channels affected by the inoperable RCS Loop Delta-T channels to be placed in the tripped condition is also conservative with respect to reactor trip and AFW pumps start. The change to the Channels to Trip and Minimum Channels Operable columns is a clarifying change to reflect the proposed changes to the action statements and identifies that the RCS Loop Delta-T does not provide a reactor trip function. Therefore, the proposed changes to the RCS Loop Delta-T function do not affect any of the accident analysis results.

The proposed changes to revise Table 3.3-4, Functional Unit 3.c.4), and to delete cycle-specific TS, TS references to RM-14A and RM-14B, and the word ''analog'' from the analog channel operation test are administrative and have no effect on plant operation.

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

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

The proposed change to the OPDT constant K6 does not affect the assumed accident initiation sequences. No new operating configuration is being imposed by the change to K6 that would create a new failure scenario. No new failure modes are being created for any plant equipment.

The proposed changes to the RCS Loop Delta-T function do not involve any physical modification to any plant system or change the methodology by which any safety-related system performs its function.

1The proposed changes to revise Table 3.3-4, Functional Unit 3.c.4), and to delete cycle-specific TS, TS references to RM-14A and RM-14B, and the word ''analog'' from the analog channel operation test are administrative, would not result in any physical alteration to any plant system, and would not be a change in the method by which any safety-related system performs its function.

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

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

The proposed change to the OPDT constant K6 will not affect any accident analysis assumptions, initial conditions, or results.

The proposed changes to the RCS Loop Delta-T function do not affect any accident analysis assumptions, initial conditions, or results.

The proposed changes to revise Table 3.3-4, Functional Unit 3.c.4), and to delete cycle-specific TS, TS references to RM-14A and RM-14B, and the word ''analog'' from the analog channel operation test are administrative and clarify the TS. These proposed changes have no effect on current operating methodologies or actions that govern plant performance.

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

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

Local Public Document Room location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407

Attorney for licensee: Christopher J. Warner, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120

NRC Project Director: Theodore R. Quay

Philadelphia Electric Company, Public Service Electric and Gas Company,Delmarva Power and Light Company, and Atlantic City Electric Company,Dockets Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station,Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: September 26, 1994

Description of amendment request: The proposed TS changes extend surveillance test intervals and allowable out-of-service times for the testing and/or repair of instrumentation that actuate the Reactor Protection System, Primary Containment Isolation, Core and Containment Cooling systems, Control Rod Blocks, Radiation Monitoring systems, and Alternate Rod Insertion/Recirculation Pump 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 changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS changes increase the STIs and AOTs for actuation instrumentation based on analyses described and justified in Licensing Topical Reports (References 2 through 8) [see licensee's September 26, 1994 application for reference information] which have been evaluated in associated Safety Evaluation Reports. These changes were incorporated into PBAPS Technical Specifications consistent with NUREG-1433. TS requirements that govern Operability or routine testing of plant instruments are not assumed to be initiators of any analyzed event because these instruments are intended to prevent, detect or mitigate accidents. Therefore, these changes will not involve an increase in the probability of occurrence of an accident previously evaluated. Additionally, these changes will not increase the consequences of an accident previously evaluated because the proposed change will not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSC are operated, maintained, modified, or inspected. The changes will not alter the operation of equipment assumed to be available for the mitigation of accidents or transients by the plant safety analysis or licensing basis. As justified in References 1 through 8, the proposed changes establish or maintain adequate assurance that components are operable when necessary for the prevention or mitigation of accidents or transients and that plant variables are maintained within limits necessary to satisfy the assumptions for initial conditions in the safety analyses. These changes establish or modify time limits allowed for operation with inoperable instrument channels based on the analyses in References 1 through 8 and will not allow continuous plant operation with plant conditions such that a single failure will result in a loss of any safety function. Therefore, these changes will not increase the consequences of an accident previously evaluated.

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

These proposed changes will not involve any physical changes to SSC, or the manner in which these SSC are operated, maintained, modified, tested, or inspected. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed TS changes increase the STIs and AOTs for actuation instrumentation based on analyses described and justified in Licensing Topical Reports (References 2 through 8) which have been evaluated in associated Safety Evaluation Reports. These changes were incorporated into PBAPS Technical Specifications consistent with NUREG-1433. These changes can be classified into one of the following three categories:

a. Changes to the minimum STIs and AOTs for the testing and/or repair of instrumentation based on the results of generic analyses in References 1 through 8;

b. Changes to conditions, required actions, and completion times needed to make PBAPS TS requirements consistent with the assumptions used in the analyses in References 1 through 8; and,

c. Changes that reformat, renumber, and/or reword existing requirements to incorporate the changes above.

All of the proposed changes will be incorporated into the PBAPS custom Technical Specifications using the same approach and specific requirements used in Reference 12.

There is no significant reduction in the margin of safety resulting from changes to the STIs and AOTs for the testing and/or repair of instrumentation based on the results of the analyses in References 1 through 8. These analyses determined that there is no significant change in the availability and/or reliability of instrumentation as a result of this change in STIs and AOTs. PECO Energy performed reviews that confirmed these analyses are applicable to PBAPS and that there would be no effect on the identification of excessive instrument setpoint drift as a result of increasing from monthly to quarterly the minimum interval between instrument functional tests. The proposed required actions ensure that actions to mitigate loss of single failure tolerance is initiated within 24 hours (12 hours for RPS) in accordance with the results of the analyses in References 1 through 8 and action to mitigate a loss of instrument function is initiated within 1 hour.

The proposed changes which replace the shutdown actions associated with inoperable instrumentation with actions to declare the supported system inoperable does not involve a reduction in a margin of safety. The proposed changes ensure that appropriate compensatory measures are taken commensurate with approved TS Actions for the affected systems and the safety analyses. In addition, the proposed changes provide the benefit of avoiding an unnecessary shutdown transient when appropriate measures are available to compensate for the inoperable instrumentation.

There is no significant reduction in the margin of safety resulting from changes that reformat, renumber, and/or reword existing requirements to incorporate the changes above.

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

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

NRC Project Director: John F. Stolz

Philadelphia Electric Company, Public Service Electric and Gas Company,Delmarva Power and Light Company, and Atlantic City Electric Company,Dockets Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station,Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: November 17, 1994

Description of amendment request: The proposed changes to the Technical Specifications (TS) are being requested to support modifications 5384 and 5386 which upgrade the Main Stack and Vent Stack Radiation Monitoring Systems.

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

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

Neither the Main Stack nor the Vent Stack Radiation Monitoring Systems serve as an initiator or contributor to any accidents previously evaluated. The systems provide indication and detection of radioactivity and effluent release in the main and vent stacks. The new systems perform the same function as the old, and have equal or better performance characteristics. Installation and operation of the new radiation monitoring systems do not degrade any active or passive equipment that responds to an accident.

The proposed increase in the surveillance test interval of the subject radiation monitoring systems from 12 to 18 months is consistent with vendor recommendations, and is based on operating experience with instrumentation of a similar design.

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

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

Both modifications replace obsolete radiation monitoring equipment and have the same failure modes as the existing equipment. The upgraded systems are considered enhancements to the existing systems and are considered neither a contributor nor initiator of any accidents previously evaluated.

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

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

Neither the accuracy nor the responsiveness of the existing radiation monitoring equipment will be degraded as a result of the installation of modifications 5384 and 5386. Revisions to the calibration and surveillance frequencies are based on vendor information and experience with instrumentation of similar design. The changes associated with setpoints and the lower limit of detection are in the conservative direction. The upgraded main stack system continues to provide a non-safety related trip signal to Group III isolation valves during purging of the containment through the SBGTS [standby gas treatment system]. The revisions to parameter descriptions and instrument designation are considered administrative.

Therefore, based on the above, 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: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105.

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

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: February 3, 1994, supplemented September 19, 1994, and November 23, 1994

Description of amendment request: The proposed amendment revises the Technical Specifications to reflect a reduction in the Reactor Coolant System flow.

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

No component modification, system realignment, or change in operations will occur which could affect the probability of any accident or transient. The proposed reduction in RCS loop and total flow rates will not change the probability of a challenge to any Engineered Safeguard Feature or other device. The consequences of previously analyzed accidents have been found to remain within acceptable licensing basis limits when the reduced flow rates are assumed. The system transient response is not affected by the initial RCS flow assumption, unless the initial assumption is so low as to impair the steady-state core cooling capability or steam generator heat transfer capability. This is clearly not the case with a 1% reduction in RCS flow. The proposed change to the wording of the parameter title on Table 3.2-1 is editorial for clarity. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Create the possibility of a new or different kind of accident.

No component modification, system realignment, or change in operating procedure will occur which could create the possibility of a new event not previously considered. The proposed reduction in RCS loop and total flow rates will not initiate any new events. Therefore, the proposed changes would not create the possibility of a different or new kind of accident.

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

The proposed decrease in RCS loop and total flow rates has been analyzed and found to have an insignificant effect on the applicable transient analyses found in the FSAR. The proposed change to the wording of the parameter title on Table 3.2-1 is editorial for clarity. Therefore, the proposed changes would not involve a significant reduction in any margin of safety.

Therefore, based on the information presented above, PSE&G has concluded there is no significant hazards consideration.

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

Local Public Document Room location: Salem Free Public library, 112 West Broadway, Salem, New Jersey 08079

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

NRC Project Director: John F. Stolz

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

Date of amendment request: January 30, 1995

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 4.6.1.2.a and associated Bases for 3/4.6.1.2 to state that Type A tests for overall integrated containment leakage rate shall be conducted in accordance with the requirements specified in Appendix J of 10 CFR 50, as modified by NRC-approved exemptions. Additionally, TS 4.6.1.2.b would be revised to eliminate the reference to the schedule contained in TS 4.6.1.2.a.

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

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because no accident initiators, conditions or assumptions are significantly affected by the proposed changes.

The proposed change would revise Technical Specification (TS) Surveillance Requirement (SR) 4.6.1.2.a to allow overall integrated containment leakage rate (Type A) testing to be scheduled in accordance with 10 CFR 50 Appendix J, as modified by approved exemptions, and would make associated administrative changes to TS SR 4.6.1.2.b and to TS Bases 3/4.6.1.2. As stated above, none of these proposed changes involve accident initiators, conditions, or assumptions.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because no accident conditions or assumptions are affected by the proposed changes.

The results of the previous Type A testing demonstrate a high degree of containment integrity. The Type B and C testing performed since the last Type A test provides confidence that the high degree of containment integrity will be maintained during the interval to the next Type A test. Therefore, the proposed changes do not alter the source term, containment isolation, or allowable releases, and will not increase the radiological consequences of a previously evaluated accident.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new or different accident initiators or assumptions are introduced by the proposed changes. The proposed changes do not affect the design or operation of any plant system, structure, or component. The proposed changes do not affect any accident initiators and are not initiators themselves. The proposed changes do not alter any accident scenarios.

3. Not involve a significant reduction in a margin of safety. The initial conditions and methodologies used in the accident analyses remain unchanged. As described above, the proposed changes do not significantly reduce or adversely affect the confidence that the present high degree of containment integrity will be maintained.

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

Local Public Document Room location: University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606.

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

NRC Project Director: Leif J. Norrholm

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

Date of amendment request: January 30, 1995

Description of amendment request: The proposed amendment would provide new Reactor Coolant Pressure Boundary (RCPB) pressure-temperature limit curves that are applicable up to 21 effective full power years (EFPY).

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

Toledo Edison had reviewed the proposed change and determined that a significant hazards consideration does not exist because operation of Davis-Besse Nuclear Power Station, Unit 1, in accordance with this change would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because: (1) revision of the pressure-temperature curves and the extended applicability of the pressurizer level/RCS pressure limit curves for periods when relief valve DH4849 is inoperable will continue to provide the same level of protection of the RCPB as was previously evaluated, and (2) the revision to License Condition 2.C(3)(d) is administrative to reflect the validity of the present analyses to 21 EFPY and (3) the revision to the Technical Specification Bases

to reflect the extension to 21 EFPY is administrative and does not affect any previously analyzed accidents.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because: (1) revision of the pressure-temperature curves and the extended applicability of the pressurizer level/RCS pressure limit curves for periods when relief valve DH4849 is inoperable will continue to provide the same level of protection of the RCPB as was previously evaluated, and (2) the revision to License Condition 2.C(3)(d) is administrative to reflect the validity of the present analyses to 21 EFPY and (3) the revision to the Technical Specification Bases to reflect the extension to 21 EFPY is administrative and does not affect any previously analyzed accidents.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because: (1) revision of the pressure-temperature curves and the extended applicability of the pressurizer level/RCS pressure limit curves will continue to provide protection against reactor vessel failure due to brittle fracture concerns under all postulated circumstances, and (2) the revision to License Condition 2.C(3)(d) is administrative to reflect the validity of the present analyses to 21 EFPY and (3) the revision to the Technical Specification Bases to reflect the extension to 21 EFPY is an administrative change and does not affect any activities or equipment in plant operation.

3. Not involve a significant reduction in a margin of safety because: (1) revision of the pressure-temperature curves and the extended applicability of the pressurizer level/RCS pressure limit curves maintains the present margin of safety from reactor vessel brittle fracture as required by 10 CFR 50, Appendix G, and (2) the revision to License Condition 2.C(3)(d) and the Bases revision are administrative and do not affect any analyses which provide the basis for the Technical Specifications.

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

Local Public Document Room location: University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606.

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

NRC Project Director: Leif J. Norrholm

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

Date of amendment request: February 14, 1995

Description of amendment request: The proposed change revises Technical Specification 4.4.D to reference the testing requirements of 10 CFR Part 50, Appendix J, and to state that the Nuclear Regulatory Commission-approved exemptions to the applicable regulatory requirements are permitted.

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:

Virginia Electric and Power Company has performed an evaluation of ... the proposed administrative Technical Specification change, in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this ... amendment request follows.

Criterion 1 - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The proposed change ... revises Technical Specification 4.4.D to reference the testing frequency requirements of 10 CFR 50 Appendix J and to state that NRC approved exemptions to the applicable regulatory requirements are permitted. The current Technical Specification requires retests in accordance with Section III.D.1(a) of Appendix J. The proposed administrative change simply includes the statement ''as modified by NRC approved exemptions.'' No new requirements are added, nor are any existing requirements deleted. Any specific changes to the requirements of Section III.D.1(a) will require a submittal from Virginia Electric and Power Company under 10 CFR 50.12 and subsequent review and approval by the NRC prior to implementation. The proposed change is stated generically to avoid the need for further Technical Specification changes if different exemptions are approved in the future.

The proposed change, in itself, does not affect reactor operations or accident analyses and has no radiological consequences. The change provides clarification so that future Technical Specifications changes will not be necessary to correspond to applicable NRC approved exemptions from the requirements of Appendix J.

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

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed Technical Specification amendment provides clarification to a specification that paraphrases a codified requirement.

Since the proposed change would not change the design, configuration or method of operation of the plant, it would not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.

The proposed Technical Specification change is administrative and clarifies the relationship between the requirements of TS 4.4.D, Appendix J, and any approved exemptions to Appendix J. It does not, in itself, change a safety limit or [a] Limiting Condition for Operation. The NRC will directly approve any proposed change or exemption to III.D.1(a) of Appendix J prior to implementation.

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

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

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

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

NRC Project Director: David B. Matthews

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.

Illinois Power Company and Soyland Power Cooperative, Inc., Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Date of amendment request: February 14, 1995

Brief description of amendment request: The amendment request proposes changes to Technical Specification 3.8.2, ''AC Sources-Shutdown;'' 3.8.5, ''DC Sources-Shutdown;'' and 3.8.8, ''Inverters-Shutdown.'' The proposed changes would revise the operability requirements for the Division 3 diesel generator and the Division 3 and 4 batteries, battery chargers, and inverters to apply only when the high pressure core spray system is required to be operable.Date of publication of individual notice in **Federal Register:** February 17, 1995 (60 FR 9412).

Expiration date of individual notice: March 20, 1995

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

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

Date of amendment request: September 8, 1994

Brief description of amendment request: The amendment request proposes changes to Technical Specification Section 3/4.9.1 to establish administrative controls to address a possible boron dilution event directly from the reactor makeup water system.

Date of publication of individual notice in **Federal Register:** March 1, 1995 (60 FR 11151).

Expiration date of individual notice: March 31, 1995

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

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.

Arizona Public Service Company, et al., Docket No. STN 50-529, Palo Verde Nuclear Generating Station, Unit 2, Maricopa County, Arizona

Date of application for amendment: November 30, 1994, as supplemented by letter dated January 27, 1995

Brief description of amendment: The amendment changed the pressurizer code safety valve lift setting from 2500 psia to 2475 psia. The lift setting is being changed to permit Unit 2 to operate with up to 1500 plugged tubes in each steam generator.

Date of issuance: March 1, 1995

Effective date: March 1, 1995

Amendment No.: 78

Facility Operating License No. NPF-74: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 4, 1995 (60 FR 496) The additional information contained in the January 27, 1995, supplemental letter was clarifying in nature and thus within the scope of the initial notice and did not affect the NRC staff's proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004

Boston Edison Company, Docket No. 50-293, Pilgrim Nuclear Power Station,Plymouth County, Massachusetts

Date of application for amendment: September 6, 1994

Brief description of amendment: The proposed amendment relocates the alarms for the drywell to suppression chamber vacuum breaker to a different annunicator panel.

Date of issuance: February 16, 1995 Effective date: To be implemented prior to startup from refueling outage ι10.

Amendment No.: 158

Facility Operating License No. DPR-35: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** October 26, 1994 (59 FR 53839) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated No significant hazards consideration comments received: No

Local Public Document Room location: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

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

Date of application for amendments: January 5, 1994, as supplemented by letters dated April 26, 1994, September 30, 1994, and January 12, 1995.

Brief description of amendments: The amendments change the Braidwood Technical Specifications to remove the requirement to verify, every 18 months, that the control room ventilation can be manually isolated.

Date of issuance: February 28, 1995

Effective date: February 28, 1995

Amendment Nos.: 60 and 60

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

Date of initial notice in **Federal Register:** January 25, 1995 (60 FR 4930). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 28, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Wilmington Township Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

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

Date of application for amendments: August 31, 1993, as supplemented July 19, 1994.

Brief description of amendments: The amendments revise the technical specifications by increasing the allowed outage time for an inoperable chiller only in MODES 1 through 4, adding an optional ACTION statement in MODES 5 and 6, and adding a surveillance requirement for the control room ventilation system.

Date of issuance: March 2, 1995

Effective date: March 2, 1995

Amendment Nos.: 70, 70, 61 and 61

Facility Operating License Nos. NPF-37, NPF-66, NPF-72 and NPF-77: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 25, 1995 (60 FR 4932). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 2, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: For Byron, the Byron Public Library, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Township Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois; Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, IllinoisDate of application for amendments: July 29, 1992, as supplemented January 14, 1993, and February 16, 1993

Brief description of amendments: Dresden and Quad Cities Technical Specification Upgrade Program. Date of issuance: February 16, 1995Effective date: Immediately, to be implemented by December 31, 1995.

Amendment Nos.: 131, 125, 152, and 148

Facility Operating License Nos. DPR-19, DPR-25, DPR-29 and DPR-30. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 23, 1993 (58 FR 34071) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 16, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: For Dresden, The Morris Public Library, 604 Liberty Street, Morris, Illinois 60450; For Quad Cities, The Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of application for amendments: July 29, 1992, as supplemented January 14, 1993, February 16, 1993 and January 27, 1995

Brief description of amendments: The July 29, 1992, application, is one of twelve applications which have been submitted by Commonwealth Edison Company (ComEd) in an effort to upgrade the existing custom Technical Specifications (TS) to the Boiling Water Reactor (BWR) Standard Technical Specifications (STS). Dresden has recently rescheduled the Unit 2 refueling outage from March 4, 1995, until June 1995. Currently, the surveillance frequency for certain Inservice Testing (IST) requirements expires on February 21, 1995. The current TSs do not make provisions for a grace period for surveillance frequencies of the IST program. In accordance with BWR STS guidance, the TSs regarding IST proposed in the July 29, 1992, application, allow the flexibility to perform these tests appropriately during refueling outages (where applicable) by providing a 25 percent extension to IST surveillance intervals. The January 27, 1995, supplement requested the staff to review and approve just that portion of the July 29, 1992, application dealing with the implementation of the IST program in Section 3.0/4.0 of the proposed TS.

Date of issuance: February 22, 1995Effective date: February 22, 1995

Amendment Nos.: 132 and 126

Facility Operating License Nos. DPR-19 and DPR-25: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 23, 1993 (58 FR 34071) The January 27, 1995, letter did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation datedFebruary 22, 1995.No significant hazards consideration comments received: No

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

Connecticut Yankee Atomic Power Company and Northeast Nuclear Energy Company, Docket Nos. 50-213 and 50-245, Haddam Neck Plant and Millstone Nuclear Power Station, Unit 1, Middlesex County and New London County, Connecticut

Date of application for amendments: October 31, 1994, as supplemented February 14, 1995.

Brief description of amendments: The amendments renew the existing license conditions for both plants to implement and maintain Integrated Implementation Schedule Program Plans (the Program Plans). The Program Plans provide a methodology to be followed for scheduling plant modifications and engineering evaluations.

Date of issuance: February 23, 1995

Effective date: February 23, 1995

Amendment Nos.: 183 for Haddam Neck, 80 for Millstone 1

Facility Operating License Nos. DPR-61 and DPR-21. Amendments revise the Licenses.

Date of initial notice in **Federal Register:** December 7, 1994 (59 FR 63117)The February 14, 1995, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 23, 1995.No significant hazards consideration comments received: No.

Local Public Document Room locations: Russell Library, 123 Broad Street, Middletown, CT 06457, for the Haddam Neck Plant, and the Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360, for Millstone Unit 1.

Consumers Power Company, Docket No. 50-255, Palisades Plant, Van Buren County, Michigan

Date of application for amendment: October 5, 1994, as supplemented February 10, 20, and 22, 1995.

Brief description of amendment: The amendment revises primary coolant system (PCS) pressure-temperature limits, power-operated relief valve setting limits, and primary coolant pump starting limits to accommodate reactor vessel fluence for an additional 4 effective full power years. The amendment also revises the emergency core cooling system technical specifications to render two high-pressure safety injection pumps incapable of injecting into the PCS when the PCS is below 300°F rather than rendering both inoperable below 260°F. In addition, it revises the pressurizer heatup to achieve consistency between design assumptions and technical specifications limits.

Date of issuance: March 2, 1995

Effective date: March 2, 1995

Amendment No.: 163

Facility Operating License No. DPR-20. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 4, 1995 (60 FR 501) The February 10, 20, and 22, 1995, submittals provided clarifyinginformation which was within the scope of the initial application and did not affect the staff's initial proposed no significant hazards consideration findings. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 2, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Van Wylen Library, Hope College, Holland, Michigan 49423.

Duke Power Company, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of application for amendments: January 10, 1994, as supplemented March 21 and September 15, 1994, and January 5, 1995

Brief description of amendments: The amendments revised Technical Specification Table 2.2-1 and TS 4.2.5 to allow a change in the method for measuring reactor coolant system (RCS) flow rate from the calorimetric heat balance method to a method based on a one-time calibration of the RCS cold leg elbow differential pressure taps.

Date of issuance: February 17, 1995

Effective date: To be implemented within 30 days from the date of issuance

Amendment Nos.: 128 and 122

Facility Operating License Nos. NPF-35 and NPF-52: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 26, 1994 (59 FR 3743) for Unit 1; and March 1, 1994 (59 FR 9785) for Unit 2

The March 21 and September 15, 1994, and January 5, 1995, letters provided additional information that did not change the initial scope of the January 10, 1994, 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 February 17, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

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

Date of amendment request: August 30, 1994

Brief description of amendment: The amendment revised the Technical Specifications to address the installation of two battery chargers on each 125 vdc power train in lieu of the ''swing'' battery charger that is currently used.

Date of issuance: February 17, 1995

Effective date: February 17, 1995

Amendment No.: 176

Facility Operating License No. DPR-51. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 17, 1995 (60 FR 3439) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 17, 1995.No significant hazards consideration comments received: No.

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

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

Date of amendment request: June 22, 1994.

Brief description of amendment: The amendment extends the allowable outage time for one inoperable train of emergency feedwater from 36 hours to 72 hours, clarifies the specifications and their associated bases, and relocates information within the specifications.

Date of issuance: March 1, 1995

Effective date: 30 days following the date of issuance.

Amendment No.: 177

Facility Operating License No. DPR-51. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 17, 1994, (59 FR 42339) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 1995.No significant hazards consideration comments received: No.

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

Entergy Operations, Inc., Docket No. 50-382, Waterford Steam ElectricStation, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: January 19, 1995

Brief description of amendment: The amendment changed the Appendix A technical specifications (TSs) by adding TS 3.0.5 and its associated Bases. This new specification will allow equipment removed from service or declared inoperable to comply with ACTIONS to be returned to service under administrative controls soley to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment.

Date of issuance: March 1, 1995

Effective date: March 1, 1995

Amendment No.: 101

Facility Operating License No. NPF-38. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 27, 1995 (60 FR 5441) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, Louisiana 70122.

Entergy Operations, Inc., Docket No. 50-382, Waterford Steam ElectricStation, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: August 11, 1994, as supplemented by letter dated December 2, 1994.

Brief description of amendment: The amendment revised the Technical Specifications for the Waterford Steam Electric Station, Unit 3, by modifying the specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to a core operating limits report for the values of those limits. These changes are in accordance with the requirements of Generic Letter 88-16.

Date of issuance: March 1, 1995

Effective date: March 1, 1995

Amendment No.: 102

Facility Operating License No. NPF-38. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 21, 1994 (59 FR 65812) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, Louisiana 70122.

Entergy Operations, Inc., Docket No. 50-382, Waterford Steam ElectricStation, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: August 19, 1994, as supplemented by letter dated October 14, 1994.

Brief description of amendment: The amendment changed the Appendix A technical specification (TSs) by removing the Limiting Condition For Operation (LCO) 3/4.3.4, the associated surveillance requirements, and Bases information from the TSs. This information and requirements will be incorporated into the Waterford 3 Updated Final Safety Analysis Report (UFSAR) and maintained under the provisions of 10 CFR 50.59.

Date of issuance: March 2, 1995

Effective date: March 2, 1995

Amendment No.: 103

Facility Operating License No. NPF-38. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 31, 1994 (59 FR 45023) The additional information contained in the supplemental letter dated October 14, 1994, was clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 2, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, Louisiana 70122.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Mississippi Power & Light Company,Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: April 21, 1993

Brief description of amendment: The amendment revised the requirement for control rod testing to increase the ''notch testing'' surveillance interval for partially withdrawn control rods from once per 7 days to once per 31 days. The change is consistent with the format and content of the Improved Standard Technical Specifications (NUREG-1434, Revision 0).

Date of issuance: February 16, 1995

Effective date: February 16, 1995

Amendment No: 115

Facility Operating License No. NPF-29. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** May 12, 1993 (58 FR 28055) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 16, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, Mississippi 39120.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Mississippi Power & Light Company,Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: July 14, 1993

Brief description of amendment: The amendment revised technical specification requirements for the hydrogen ignition system (HIS). The amendment also removed several tables related to the HIS in accordance with guidance contained in Generic Letter 91-08, ''Removal of Component Lists From Technical Specifications.''

Date of issuance: February 16, 1995

Effective date: February 16, 1995

Amendment No: 116

Facility Operating License No. NPF-29. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** September 1, 1993 (58 FR 46232) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 16, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Judge George W. Armstrong Library, 220 S. Commerce at Washington, Natchez, Mississippi 39120.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Mississippi Power & Light Company,Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: August 11, 1993

Brief description of amendment: The amendment deleted the requirements of Limiting Condition for Operation (LCO) 3.3.3.9 and Surveillance Requirement 4.3.3.9 related to loose-part detection instrumentation. The deleted requirements will be relocated to documents that are controlled by the licensee under the provisions of 10 CFR 50.59. The change is consistent with the format and content of the Improved Standard Technical Specifications (NUREG-1434, Revision 0).

Date of issuance: February 16, 1995

Effective date: February 16, 1995

Amendment No: 117

Facility Operating License No. NPF-29. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** September 1, 1993 (58 FR 46232) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 16, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, Mississippi 39120.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Mississippi Power & Light Company,Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: August 11, 1993

Brief description of amendment: The amendment deleted certain accident monitoring instruments from Technical Specification Table 3.3.7.5-1 ''Accident Monitoring Instrumentation'' and deleted the corresponding Surveillance Requirements from Table 4.3.7.5-1, ''Accident Monitoring Instrumentation Surveillance Requirements.'' The deleted requirements will be relocated to documents that are controlled by the licensee under the provisions of 10 CFR 50.59. The change is consistent with the format and content of the Improved Standard Technical Specifications (NUREG-1434, Revision 0).

Date of issuance: February 16, 1995

Effective date: February 16, 1995

Amendment No: 118

Facility Operating License No. NPF-29. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** September 1, 1993 (58 FR 46234) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 16, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, Mississippi 39120.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Mississippi Power & Light Company, Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: October 22, 1993, as supplemented by letters dated February 10, and 14, 1995.

Brief description of amendment: The amendment modified the testing frequencies for the drywell bypass test and the airlock test, relocated certain drywell airlock tests from the technical specifications to administrative procedures, and incorporates various improvements from the Improved Standard Technical Specifications (NUREG-1434, Revision 0).

Date of issuance: February 16, 1995

Effective date: February 16, 1995

Amendment No: 119

Facility Operating License No. NPF-29. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** December 8, 1993 (58 FR 64607) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 16, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, Mississippi 39120.

GPU Nuclear Corporation, et al., Docket No. 50-289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

Date of application for amendment: May 17, 1993, as supplemented on December 23, 1994

Brief description of amendment: The amendment changes the action statement for inoperable degraded grid and loss of voltage relays and their associated auxiliary relays and timers.

Date of issuance: January 31, 1995

Effective date: January 31, 1995

Amendment No.: 193

Facility Operating License No. DPR-50. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 10, 1993 (58 FR 59750). The December 23, 1994, letter provided additional information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated January 31, 1995. No significant hazards consideration comments received: No.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105. The above Notice was to be published in the **Federal Register** of February 15, 1995. The notice that was inadvertently published at 60 FR 8762 relates to a licensing action which has not been completed.

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

Date of amendment request: November 7, 1994, as supplemented by letters dated December 20, 1994, and January 23, 1995.

Brief description of amendments: The amendments changed the number of standby diesel generators (SDGs) (emergency power source) required to be operable during Mode 6 with greater than or equal to 23 feet of water above the reactor vessel flange, from two to one. The amendment also allows limited substitution of an alternate onsite emergency power source for one of the two required SDGs, in Mode 5, and in Mode 6 with less than 23 feet of water. In addition, certain system specifications that are affected by the changes for the emergency power source were also changed.

Date of issuance: February 14, 1995

Effective date: February 14, 1995, to be implemented within 31 days of issuance.

Amendment Nos.: Unit 1 - Amendment No. 34; Unit 2 - Amendment No. 20

Facility Operating License Nos. NPF-76 and NPF-80. The amendments revised the Technical

Specifications.Public comments requested as to proposed no significant hazards consideration: Yes (60 FR 5739, dated January 30, 1995). 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 March 1, 1995, but stated 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 final determination of no significant hazards consideration is contained in a Safety Evaluation dated February 14, 1995.

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

IES Utilities Inc., Docket No. 50-331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: August 15, 1994, as supplemented on December 21, 1994, and January 20, 1995. The licensee's submittals of December 21, 1994, and January 20, 1995, provided clarification and did not change the original no significant hazards consideration.

Brief description of amendment: The proposed amendment would revise the Technical Specifications by increasing the allowable main steam isolation valve (MSIV) leakage and deleting the requirements applicable to the MSIV leakage control system.

Date of issuance: February 22, 1995

Effective date: February 22, 1995 and to be implemented within 90 days.

Amendment No.: 207

Facility Operating License No. DPR-49. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** September 14, 1994 (59 FR 47169) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 22, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, S. E., Cedar Rapids, Iowa 52401.

Illinois Power Company and Soyland Power Cooperative, Inc., Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Date of application for amendment: August 12, 1994, as supplemented on October 14, 1994 and February 6, 1995.

Brief description of amendment: The amendment modifies Clinton Power Station Technical Specification 3.6.5.1, ''Drywell,'' to permit a one-time only change to forego performance of the drywell bypass leakage rate test during the fifth refueling outage scheduled to begin in March 1995.

Date of issuance: March 1, 1995

Effective date: March 1, 1995

Amendment No.: 96

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

Date of initial notice in **Federal Register:** September 28, 1994 (59 FR 49427). The October 14, 1994, and February 6, 1995, submittals consisted of revisions and clarifications which did not change the staff's initial proposed no significant hazards consideration determination or expand the scope of the original notice.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 1995. No significant hazards consideration comments received: No

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

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

Date of application for amendments: November 18, 1994

Brief description of amendments: The amendments revise Technical Specification 4.0.5 to delete the wording ''except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).'' This change allows the licensee to implement certain 10 CFR 50.55a relief requests while the relief requests are being reviewed by the NRC at the beginning of an updated interval.

Date of issuance: February 23, 1995

Effective date: February 23, 1995

Amendment Nos.: 190/176

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

Date of initial notice in **Federal Register:** December 21, 1994 (59 FR 65817) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 23, 1995. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 18, 1994

Brief description of amendment: The amendment modifies the operability requirements for the fuel building exhaust filter system. The amendment will result in modifications to the applicability, surveillance requirement, and bases sections of Technical Specification 3/4.9.12, ''Fuel Building Exhaust Filter System.''

Date of issuance: February 22, 1995

Effective date: As of the date of issuance to be implemented within30 days.

Amendment No.: 105

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

Date of initial notice in **Federal Register:** June 22, 1994 (59 FR 32234) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 22, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

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

Date of application for amendments: July 9, 1992

Brief description of amendments: The amendments extend the operating licenses for the Diablo Canyon Nuclear Power Plant, Units 1 and 2 to recover or recapture the construction period of the reactors. Specifically, the amendments extend the expiration date of the Unit 1 license from April 23, 2008, to September 22, 2021, and the expiration date of the Unit 2 license from December 9, 2010, to April 26, 2025.

Date of issuance: March 1, 1995

Effective date: March 1, 1995

Amendment Nos.: 97 and 96

Facility Operating License Nos. DPR-80 and DPR-82: The amendments revised the license.

Date of initial notice in **Federal Register:** July 22, 1992 (57 FR 32575)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 1, 1995.No significant hazards consideration comments received: Yes. Comments from the San Luis Obispo Mothers for Peace (MFP) and their contentions were admitted into this proceeding. These contentions concern the adequacy of the licensee's maintenance and surveillance program and interim corrective actions in lieu of Thermo-Lag. The Atomic Safety and Licensing Board, in its initial decision dated November 4, 1994 (LBP-94-35), authorized the staff to extend the DCPP operating license expiration dates. Because a hearing was held prior to license issuance, the staff does not need to make a final no significant hazards consideration determination.

Local Public Document Room location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407

Pennsylvania Power and Light Company, Docket No. 50-387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of application for amendment: July 27, 1994, as supplemented October 27, 1994 and February 3, 1995

Brief description of amendment: The amendment raises the authorized Power Level from 3293 MWt to a new limit of 3441 MWt.

Date of issuance: February 22, 1995

Effective date: As of date of issuance and is to be implemented prior to startup in Cycle 9, currently scheduled to occur in May 1995.

Amendment No.: 143

Facility Operating License No. NPF-14: This amendment revised the Technical Specifications and license.

Date of initial notice in **Federal Register:** September 14, 1994 (59 FR 47171) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 22, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendments: June 23, 1994

Brief description of amendments: The amendment revises Technical Specification 4.0.5, which provides the requirements for inservice inspection and testing of ASME Code components, to conform to Standard Technical Specifications (NUREG-1433).

Date of issuance: February 28, 1995

Effective date: February 28, 1995

Amendment Nos.: 144 and 113

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

Date of initial notice in **Federal Register:** August 3, 1994 (59 FR 39595) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 28, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendments: October 28, 1994, and supplemented by letter dated December 29, 1994

Brief description of amendments: These amendments change the Technical Specifications (TS) for the two units by adding reference ι20 (Unit 1) and reference ι18 (Unit 2) to Section 6.9.3.2 as ''PL-NF-90-001, Supplement 1, 'Application of Reactor Analysis Methods for BWR Design and Analysis: Loss of Feedwater Heating Changes and Use of RETRAN MOD 5.1,' September 1994.'' These additions reflect changes to the methodology that the licensee is using to perform its nuclear fuel reload analysis for the two units.

Date of issuance: February 28, 1995

Effective date: February 28, 1995

Amendment Nos.: 145 and 114

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

Date of initial notice in **Federal Register:** December 21, 1994 (59 FR 65819) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 28, 1995.No significant hazards consideration comments received: No

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

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of application for amendments: August 31, 1994

Brief description of amendments: These amendments address Section 5, ''Remove Temperature Requirement for Operational Condition 5 (TSCR 94-44-0), by revising TS Table 1.2 and TS Bases 3/4.9.11 to remove the average reactor coolant temperature requirement in Operational Condition (OPCON) 5, Refueling.

Date of issuance: January 27, 1995

Effective date: January 27, 1995Amendment Nos. 88 and 50

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

Date of initial notice in **Federal Register:** November 9, 1994 (59 FR 55884) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 27, 1995.No significant hazards consideration comments received: No

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

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 application for amendment: November 16, 1994

Brief description of amendment: The amendment revises Technical Specifications Section 3.10.8 and the associated Bases, to reduce the maximum allowable control rod drop time from 2.4 to 1.8 seconds.

Date of issuance: February 21, 1995

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

Amendment No.: 160

Facility Operating License No. DPR-64: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 20, 1995 (60 FR 4203) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 21, 1995.No significant hazards consideration comments received: No

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

Saxton Nuclear Experimental Corporation, Docket No. 50-146, Saxton Nuclear Reactor Facility

Date of application for amendment: August 8, 1994, as supplemented on October 28, 1994, and January 12, 1995.

Brief description of amendment: The amendment adds characterization as an authorized activity at Saxton and improves the wording of the technical specifications.

Date of issuance: February 22, 1995

Effective date: February 22, 1995

Amendment No.: 12Amended Facility License No. DPR-4: Amendment changed the Technical Specifications

Date of initial notice in **Federal Register:** November 9, 1994. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 22, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Saxton Community Library, 911 Church Street, Saxton, Pennsylvania 16678

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: December 30, 1993, as supplemented by letters dated June 3, 1994, August 25, 1994, and January 3, 19, and 30, 1995.

Brief description of amendments: These amendments revise TS 3.9.4, ''Containment Building Penetrations,'' and the associated bases to allow both doors of the containment personnel airlock to be open at the same time during refueling operations provided certain conditions are met.

Date of issuance: February 28, 1995

Effective date: February 28, 1995

Amendment Nos.: 117 and 106

Facility Operating License Nos. NPF-10 and NPF-15: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** September 28, 1994 (59 FR 49434). The additional information contained in the January 3, 19, and 30, 1995, letters were clarifying in nature, within the scope of the initial notice and did not affect the NRC staff's proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 28, 1995.No significant hazards consideration comments received: No.

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

Southern Nuclear Operating Company, Inc., Docket Nos. 50-348 and 50-364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama.

Date of amendments request: December 19, 1994

Brief description of amendments: The amendments to Technical Specifications include: (1) a revision in Table 3.7-3 to the main steam safety valve (MSSV) setpoint tolerance from plus or minus 1 percent to plus or minus 3 percent, (2) modification of the bases to 3/4.7.1.1 to increase the relieving capacity of the MSSVs to at least 12,984,660 pounds per hour which corresponds to approximately 112 percent of total secondary steam flow at 100 percent rated thermal power, (3) modifications to Table 3.7-1 to reduce the allowable power range neutron flux high setpoints for multiple inoperable steam generator safety valves, and (4) an editorial correction to Bases 3/4.7.1.2 to indicate required auxiliary feedwater flow at ''1133 psia'' rather than ''1133 psig.''

Date of issuance March 1, 1995

Effective date: March 1, 1995

Amendment Nos.: 112 and 103

Facility Operating License Nos. NPF-2 and NPF-8. Amendments revise the Technical Specifications.

Date of initial notice in **Federal Register:** January 4, 1995 (60 FR 505) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 1, 1995No significant hazards consideration comments received: No

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 1, 1995No significant hazards consideration comments received: No

Tennessee Valley Authority, Docket Nos. 50-259, 50-260, and 50-296, Browns Ferry Nuclear Plant, Units 1, 2, and 3, Limestone County, Alabama

Date of application for amendments: September 29, 1993

Brief description of amendment: The proposed changes increase the amount of boron required in the standby liquid control system.

Date of issuance: February 28, 1995

Effective date: February 28, 1995

Amendment Nos.: 217, 233 and 191

Facility Operating License Nos. DPR-33, DPR-52 and DPR-68: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 8, 1994 (59 FR 29635) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 28, 1995.No significant hazards consideration comments received: None

Local Public Document Room location: Athens Public library, South Street, Athens, Alabama 35611

Tennessee Valley Authority, Docket Nos. 50-259, 50-260 and 50-296, Browns Ferry Nuclear Plant, Units 1, 2, and 3, Limestone County, Alabama

Date of application for amendments: September 30, 1993 (TS 336)

Brief description of amendment: The proposed changes revise and clarify the spent fuel pool water level, temperature, sampling, and analysis surveillance requirements.

Date of issuance: March 2, 1995

Effective date: March 2, 1995

Amendment Nos.: 218, 334 and 192

Facility Operating License Nos. DPR-33, DPR-52 and DPR-68: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 22, 1993 (58 FR 67862) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 2, 1995.No significant hazards consideration comments received: None

Local Public Document Room location: Athens Public library, South Street, Athens, Alabama 35611

Tennessee Valley Authority, Docket Nos. 50-259, 50-260, and 50-296, Browns Ferry Nuclear Plant, Units 1, 2, and 3, Limestone County, Alabama

Date of application for amendments: March 31, 1994

Brief description of amendment: For Browns Ferry Units 1 and 3, the proposed changes provide for operation in the extended load line limit region and revised rod block monitor operability requirements. For all three Browns Ferry units, the changes delete a obsolete value for rated loop recirculation flow rate, relocate cycle-specific equations to the Core Operating Limits report, and provide other miscellaneous changes.

Date of issuance: February 24, 1995

Effective date: February 24, 1995

Amendment Nos.: 216, 232, 190

Facility Operating License Nos. DPR-33, DPR-52 and DPR-68: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** September 28, 1994 (59 FR 49437) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 24, 1995.No significant hazards consideration comments received: None

Local Public Document Room location: Athens Public library, South Street, Athens, Alabama 35611

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: October 7, 1994

Brief description of amendment: Eliminates redundancy in system leakage test requirements by revising TS 3/4.5.2 and its associated basis for the Emergency Core Cooling System and TS 3/4.6.2 and its associated basis for the Containment Spray System.

Date of issuance: February 27, 1995

Effective date: February 27, 1995 and to be implemented within 90 days.

Amendment No. 195

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

Date of initial notice in **Federal Register:** November 9, 1994 (59 FR 55893) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 27, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606.

TU Electric Company, Docket Nos. 50-445 and 50-446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: November 18, 1994 (published in Federal Register as November 11, 1994)

Brief description of amendments: The proposed amendments would provide for cycle-specific allowances to account for increases in the Heat Flux Hot Channel Factor between monthly surveillances.

Date of issuance: March 1, 1995

Effective date: March 1, 1995, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 1 - Amendment No. 34; Unit 2 - Amendment No. 20

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

Date of initial notice in **Federal Register:** December 7, 1994 (59 FR 63127) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 1, 1995.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 31, 1994

Brief description of amendment: The amendment modifies the Techical Specifications (TS) to (1) add two action statements that would provide allowed outage times for either one or both of the scram discharge volume (SDV) vent or drain valves less stringent than the current requirements of TS 3.0.3., and (2) change the surveillance requirements for the SDV vent and drain valves to conduct the testing during shutdown conditions rather than at power as currently required.

Date of issuance: February 27, 1995

Effective date: February 27, 1995

Amendment No.: 134

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

Date of initial notice in **Federal Register:** December 21, 1994 (59 FR 65828) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 27, 1995.No significant hazards consideration comments received: No

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

Wisconsin Public Service Corporation, Docket No. 50-305, Kewaunee NuclearPower Plant, Kewaunee County, Wisconsin

Date of application for amendment: February 23, 1994

Brief description of amendment: The amendment revises Kewaunee Nuclear Power Plant (KNPP) Technical Specification (TS) 6.8.c by removing the requirement to conduct a biennial review of plant procedures in accordance with American National Standards Institute (ANSI) N18.7-1976, Section 5.2.15. Alternate programs that are described in the KNPP Operational Quality Assurance Program Description (OQAPD) will be used to ensure that procedures are reviewed and maintained current.

Date of issuance: February 23, 1995

Effective date: February 23, 1995 and to be implemented within 30 days.

Amendment No.: 115

Facility Operating License No. DPR-43. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 30, 1994 (59 FR 14903) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 23, 1995.No significant hazards consideration comments received: None.

Local Public Document Room location: University of Wisconsin Library Learning Center, 2420 Nicolet Drive, Green Bay, Wisconsin 54301.

Notice Of Issuance Of Amendments To Facility Operating Licenses And Final Determination Of No Significant Hazards Consideration And Opportunity For A Hearing (Exigent Public Announcement Or Emergency Circumstances)

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

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

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

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

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

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

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By April 14, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

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

Pennsylvania Power and Light Company, Docket No. 50-388, SusquehannaSteam Electric Station, Unit 2, Luzerne County, Pennsylvania

Date of application for amendment: February 7, 1995

Brief description of amendment: The amendment changed the Technical Specifications to allow continued operation with one neutron flux monitor system channel (≥B'' channel) inoperable and should the remaining channel become inoperable to allow continued plant operation for 7 days to restore one of the two inoperable channels.

Date of issuance: March 1, 1995

Effective date: March 1, 1995

Amendment No.: 115

Facility Operating License No. NPF-22: Amendment revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: No. On February 8, 1995, the staff issued a Notice of Enforcement Discretion, which was immediately effective and remained in effect until this amendment was issued.

The Commission's related evaluation of the amendments, finding of emergency circumstances, consultation with the Commonwealth of Pennsylvania and final no significant hazards considerations determination are contained in a Safety Evaluation dated March 1, 1995.

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

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18071.

NRC Project Director: John F. Stolz

Dated at Rockville, Maryland, this 8th day of March 1995.

For the Nuclear Regulatory Commission

Elinor G. Adensam,

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

[Doc. 95-6207 Filed 3-14-95; 8:45 am]

BILLING CODE 7590-01-F

[Docket No. 50–346]

Toledo Edison Company, et al.; Notice of Withdrawal of Applications for Amendments to Facility Operating License

The U.S. Nuclear Regulatory Commission (the Commission) has granted the request of the Toledo Edison Company, Centerior Service Company, and the Cleveland Electric Illuminating Company (the licensees) to withdraw its March 13, 1992, September 11, 1992, and February 17, 1993, applications for proposed amendments to Facility Operating License No. NPF–3 for the Davis-Besse Nuclear Power Station, Unit No. 1, located in Ottawa County, Ohio.

The proposed amendments would have revised the facility technical specifications by changing the venting requirements for the Reactor Coolant System, deleting figures in Section 5.1, ''Design Features—Site,'' and revising the Safety Features Actuation System and Steam and Feedwater Rupture Control System Instrumentation Setpoints.

The Commission had previously issued Notice of Consideration of Issuance of Amendment published in the **Federal Register** on March 23, 1992 (57 FR 10050), for the March 13, 1992, application; January 6, 1993 (58 FR 600) for the September 11, 1992, application, and June 23, 1993 (58 FR 34096), for the February 23, 1993, application. However, by letter dated February 10, 1995, the licensee withdrew the proposed changes.

For further details with respect to this action, see the applications for amendment dated March 13, 1992, September 11, 1992, and February 17, 1993, and the licensee's letter dated February 10, 1995, which withdrew the applications for license amendments. The above documents are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606.

Dated at Rockville, Maryland, this 9th day of March, 1995.

For the Nuclear Regulatory Commission.

Jon B. Hopkins,

Senior Project Manager, Project Directorate III–3, Division of Reactor Projects III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 95–6340 Filed 3–14–95; 8:45 am]

BILLING CODE 7590–01–M

OFFICE OF THE UNITED STATES TRADE REPRESENTATIVE

Notice of Meeting of the Industry Policy Advisory Committee

AGENCY: Office of the United States Trade Representative.

AGENCY: Notice that the March 22, 1995 meeting of the Industry Policy Advisory Committee will be held from 9:30 a.m. to 2:30 p.m. The meeting will be closed to the public from 9:30 to 1:00 p.m. The meeting will be open to the public from 1:00 p.m. to 2:30 p.m.

SUMMARY: The Industry Policy Advisory Committee will hold a meeting on March 22, 1995 from 9:30 a.m. to 2:30 p.m. The meeting will be closed to the public from 9:30 to 1:00 p.m. The meeting will include a review and discussion of current issues which influence U.S. trade policy. Pursuant to Section 2155(f)(2) of Title 19 of the United States Code, I have determined that this portion of the meeting will be concerned with matters the disclosure of which would seriously compromise the development by the United States Government of trade policy, priorities, negotiating objectives or bargaining positions with respect to the operation of any trade agreement and other matters arising in connection with the development, implementation and administration of the trade policy of the United States. The meeting will be open to the public and press from 1:00 p.m. to 2:30 p.m. when trade policy issues will be discussed. Attendance during this part of the meeting is for observation only. Individuals who are not members of the committee will not be invited to comment.

DATES: The meeting is scheduled for March 22, 1995, unless otherwise notified.

ADDRESSES The meeting will be held at the Madison Hotel, located at 15th and M streets, NW., Washington, DC., unless otherwise notified.

FOR FURTHER INFORMATION CONTACT: Michaelle Burstin, Director of Public Liaison, Office of the United States Trade Representative, (202) 395–6120.

Michael Kantor,

United States Trade Representatives.

[FR Doc. 95–6316 Filed 3–14–95; 8:45 am]

BILLING CODE 3190–01–M

POSTAL RATE COMMISSION

Notice of Facility Visit

March 10, 1995.

Members of the Commission and its advisory staff will visit the offices of