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 the amendment is issued before the expiration of the 30-day hearing period, the Commission will make a final determination on the issue of no significant hazards consideration. If a hearing is requested, the final determination will serve to decide when the hearing is held.

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, NW, Washington, DC 20036–5869, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding 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).

For further details with respect to this action, see the application for amendment dated August 28, 1998, 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, located at the Wharton County Junior College, J.M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

Dated at Rockville, Maryland, this 2nd day of September 1998.

For the Nuclear Regulatory Commission.

Thomas W. Alexion,

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

[FR Doc. 98–24127 Filed 9–8–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Weeks of September 7, 14, 21, and 28, 1998.*

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of September 7

Thursday, September 10

3:30 p.m. Affirmative Session (Public Meeting) (if needed)

Week of September 14—Tentative

Tuesday, September 15

2:00 p.m. Briefing by Reactor Vendors Owners Groups (Public Meeting) (Contact: Bryan Sheron, 301–415–1274)

3:30 p.m. Affirmation Session (Public Meeting) (if needed)

Wednesday, September 16

10:00 a.m. Briefing on Investigative Matters (Closed—Ex. 5 and 7)

Week of September 21—Tentative

There are no meetings the week of September 21.

Week of September 28—Tentative

There are no meetings the week of September 28.

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*The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292.

CONTACT PERSON FOR MORE INFORMATION: Bill Hill (301) 415–1661.

The NRC Commission Meeting Schedule can be found on the Internet at:

http://www.nrc.gov/SECY/smj/schedule.htm

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, DC 20555 (301–415–1661). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@nrc.gov or dkw@nrc.gov.

Dated: September 4, 1998.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 98–24354 Filed 9–4–98; 3:48 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the

Commission that such amendment involves no significant hazards consideration, notwithstanding the 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 August 17, 1998, through August 28, 1998. The last biweekly notice was published on August 26, 1998 (63 FR 45521).

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

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

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

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

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

By October 9, 1998, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

Date of amendment request: August 17, 1998.

Description of amendment request: The Carolina Power & Light Company, licensee for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2, proposed amendments to the Technical Specifications (TS) to revise the requirement that the operations manager hold or has held a senior reactor operator (SRO) license. The proposed revision would require that either the operations manager or assistant operations manager hold an SRO license.

The licensee has concluded that the proposed license amendments do not involve a Significant Hazards Consideration. In support of this determination, an evaluation of each of the three standards set forth in 10 CFR 50.92 is provided below.

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 operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed license amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated. The change to Technical Specification 5.2.2.f to require the operations manager or assistant operations manager to hold an SRO license is administrative in nature and does not directly affect plant operations. The change does not physically alter the facility in any manner and, as such, does not affect the means in which any safety-related system performs its intended safety function.

2. Would operation of the facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed license amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. As stated above, the proposed change is administrative in nature. It does not involve physical alterations of the plant configuration or changes in setpoints or operating parameters. Therefore, there is no possibility of creating a new or different kind of accident.

3. Would operation of the facility in accordance with the proposed amendment involve a significant reduction in a margin of safety?

The proposed license amendments do not involve a significant reduction in a margin of safety. The proposed change to Technical Specification 5.2.2.f, requiring the operations manager or assistant operations manager to hold an SRO license is consistent with (1) 10 CFR 50.54(l), which requires individuals responsible for directing the licensed activities of licensed operators to hold an SRO license, (2) the previously approved wording of Revision 1 of NUREG–1433, ''Standard Technical Specifications General Electric Plants, BWR/4,'' and Technical Specification Traveler Form (TSTF) 65, Revision 1, and (3) the intent of ANSI–N18.1.–1971, ''Selection and Training of Nuclear Power Plant Personnel.'' Therefore, the proposed change does not represent a reduction in the margin of safety.

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

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

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

NRC Project Director: Pao-Tsin Kuo (Acting).

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, Illinois

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

Date of application for amendment request: August 14, 1998.

Description of amendment request: The proposed amendments would change the Dresden, Quad Cities, and LaSalle Technical Specifications (TS) to reflect the use of Siemens Power Corporation (SPC) ATRIUM–9B fuel. Specifically the proposed amendments incorporate the following into the TS: (a) new methodologies that will enhance operational flexibility and reduce the likelihood of future plant derates, (b) administrative changes that eliminate the cycle specific implementation of ATRIUM–9B fuel and adopt Improved Standard Technical Specification language where appropriate, and (c) changes to the Minimum Critical Power Ratio (MCPR). This amendment request supersedes in its entirety a letter from J. Hosmer (ComEd) to U.S. NRC, ''Technical Specification Changes for Transition to Siemens Power Corporation ATRIUM–9B Fuel,'' dated August 29, 1997 (63 FR 2274).

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 evaluated.

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits have been established consistent with NRC approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. These changes do not affect the operability of plant systems, nor do they compromise any fuel performance limits.

a. Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

The Reference 1 methodology to be added to the Technical Specifications is used as part of the LOCA analysis and does not introduce physical changes to the plant. The Reference 1 revised jet pump model changes the calculational behavior of the jet pump under reversed drive flow conditions. The revised jet pump model methodology makes the LOCA model behave more realistically and calculates small break LOCA PCTs that are comparable to the large break LOCA results. Therefore, this change only affects the methodology for analyzing the LOCA event and determining the protective APLHGR limits. The Technical Specification requirements for monitoring APLHGR are not affected by this change. The revised method will result in higher APLHGR limits, thus the SPC fuel will be allowed to operate at higher nodal powers. The approved methodology, however, still protects the fuel performance limits specified by 10 CFR 50.46. Therefore, the probability or consequences of an accident previously evaluated will not change.

b. Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

The probability or consequences of a previously evaluated accident are not increased by adding Reference 3 to Section 6.9.A.6.b of the Quad Cities Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications. Reference 3 determines the additive constants and the associated uncertainty for application of the ANFB correlation to the coresident GE fuel. Therefore, it provides data that is used in the determination of the MCPR Safety Limit. This approved methodology for applying the ANFB critical power correlation to the GE fuel will protect the fuel from boiling transition. Operational MCPR limits will also be applied to ensure that the MCPR Safety Limit is protected during all modes of operation and anticipated operational occurrences. Because Reference 3 contains conservative methods and calculations and because the operability of plant systems designed to mitigate any consequences of accidents have not changed, the probability or consequences of an accident previously evaluated will not increase.

c. Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The probability or consequences of a previously evaluated accident are not increased by adding Reference 7 to Section 6.9.A.6.b of the Quad Cities and Dresden Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications. Approval of Reference 7 (Reference 20) documents the additive constant uncertainty for the SPC ATRIUM–9B fuel design with an internal water channel. This methodology is used to determine an input to the MCPR Safety Limit calculations, which ensures that at least 99.9% of the fuel rods avoid transition boiling during normal operation as well as anticipated operational occurrences. This change does not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. This methodology for determining the ATRIUM–9B additive constant uncertainty for the MCPR Safety Limit calculation will continue to support protecting the fuel from boiling transition. Operational MCPR limits will be applied to ensure the MCPR Safety Limit is not violated during all modes of operation and anticipated operational occurrences. Therefore, no individual precursors of an accident are affected and the operability of plant systems designed to mitigate the probability or the consequences of an accident previously evaluated is not affected by these changes.

d. Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2, Dresden Unit 3, and LaSalle Units 1 and 2)

Changing the MCPR Safety Limit at Quad Cities Units 1 and 2, Dresden Unit 3, and LaSalle Units 1 and 2 will not increase the probability or the consequences of an accident previously evaluated. This change implements the MCPR Safety Limits resulting from the SPC ANFB critical power correlation methodology using the ATRIUM–9B additive constant uncertainty resulting from approval of Reference 7 (Reference 20). The MCPR Safety Limits for Quad Cities Units 1 and 2, Dresden Unit 3, and LaSalle Units 1 and 2 are anticipated to be conservative and acceptable for future cycles. Cycle specific MCPR Safety Limit calculations will be performed, consistent with SPC's approved methodology, to confirm the appropriateness of the MCPR Safety Limit. Additionally, operational MCPR limits will be applied that will ensure the MCPR Safety Limit is not violated during all modes of operation and anticipated operational occurrences. The MCPR Safety Limits are being set at the CPR value where less than 0.1% of the rods in the core are expected to experience boiling transition. These Safety Limits are expected to be applicable for future cycles of ATRIUM–9B. Therefore the probability or consequences of an accident will not increase.

e. Removal of Footnotes Limiting Operation with ATRIUM–9B Fuel Reloads (Quad Cities Unit 2 and Dresden Units 2 and 3)

The removal of footnotes from the Quad Cities and Dresden Technical Specifications does not involve any significant increase in the probability or consequences of an accident previously evaluated. The footnotes were added to clarify that cycle specific methods were used until the generic methodology was approved by the NRC. Since the NRC has approved SPC's generic methodology for application of the ANFB correlation to the coresident GE fuel (Reference 3) and SPC has addressed the concerns regarding the database used to calculate the ATRIUM–9B additive constant uncertainties (Reference 7), the footnotes are no longer necessary. The removal of the Unit 2 specific ''a'' pages, 2–1a and B2–3a, in the Quad Cities Technical Specifications is justified by the removal of the footnotes. Therefore, removing these footnotes and ''a'' pages does not require any physical plant modifications, nor does it physically affect any plant components or entail changes in plant operation. Therefore, the probability or consequences of an accident previously evaluated are not expected to increase.

f. Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The revision to the Section 3 Technical Specification description of the APLHGR limits has no implications on accident analysis or plant operations. The purpose of the revision is to allow flexibility for the MAPLHGR limits and their exposure basis to be specified in the COLR and to establish consistency with approved methodologies currently utilized by Siemens Power Corporation, which calculate MAPLHGR limits based on bundle or planar average exposures. This revision also provides for consistency in the APLHGR limit Technical Specification wording between the ComEd BWRs. The revision to the 3.11.D SLHGR Technical Specification for Dresden also has no implications on accident analysis or plant operations. The purpose of this revision is to allow flexibility for the LHGR limits and their exposure basis to be specified in the COLR. This revision makes the Dresden LHGR definition consistent with NUREG 1433/1434, Revision 1 wording. The definition of the Average Planar Exposure is deleted, because the exposure basis of the APLHGR and LHGR is being removed. Therefore, no plant equipment or processes are affected by this change. Thus, there is no alteration in the probability or consequences of an accident previously evaluated.

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

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications to the plant configuration, including changes in allowable modes of operation. This Technical Specification submittal does not involve any modifications to the plant configuration or allowable modes of operation. No new precursors of an accident are created and no new or different kinds of accidents are created. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

a. Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

The revised jet pump model methodology will be used to analyze the LOCA for LaSalle Units 1 and 2, and does not introduce any physical changes to the plant or the processes used to operate the plant. This change only affects the methods used to analyze the LOCA event and determine the MAPLHGR limits. Therefore, the possibility of a new or different kind of accident is not created.

b. Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

Addition of the generic methodology for the application of the ANFB critical power correlation to GE fuel in Section 6.9.A.6.b of the Quad Cities Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications does not introduce any physical changes to the plant, the processes used to operate the plant, or allowable modes of operation. This change only involves adding an NRC approved methodology, which is used to determine the additive constants and additive constant uncertainty for GE fuel, to Section 6 of the Technical Specifications. Therefore, no new precursors of an accident are created and no new or different kinds of accidents are created.

c. Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

Addition of the Reference 7 methodology to Section 6.9.A.6.b of the Quad Cities and Dresden Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications will not create the possibility of a new or different kind of accident from any accident previously evaluated. This methodology describes the calculation of an input to the MCPR Safety Limit—the ATRIUM–9B additive constant uncertainty. This change does not introduce any physical changes to the plant, the processes used to operate the plant, or allowable modes of operation. Therefore, no new precursors of an accident are created and no new or different kinds of accidents are created.

d. Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2, Dresden Unit 3, and LaSalle Units 1 and 2)

Changing the MCPR Safety Limit will not create the possibility of a new accident from an accident previously evaluated. This change will not alter or add any new equipment or change modes of operation. The MCPR Safety Limit is established to ensure that 99.9% of the rods avoid boiling transition.

The MCPR Safety Limit is changing for Quad Cities, Dresden Unit 3 and LaSalle due to the revised ATRIUM–9B additive constants and the ATRIUM–9B additive constant uncertainty resulting from approval of Reference 7 (Reference 20). The new MCPR Safety Limit for Quad Cities Units 1 and 2, Dresden Unit 3, and LaSalle Units 1 and 2 are greater than the current values at Quad Cities Units 1 and 2, Dresden Unit 3, and LaSalle Units 1 and 2 and are being increased now in anticipation of bounding future reloads of ATRIUM–9B. This change does not introduce any physical changes to the plant, the processes used to operate the plant, or allowable modes of operation. Therefore, no new accidents are created that are different from any accident previously evaluated.

e. Removal of Footnotes Limiting Operation with ATRIUM–9B Fuel Reloads (Quad Cities Unit 2 and Dresden Units 2 and 3)

The removal of the footnotes from the Quad Cities and Dresden Technical Specifications does not create a new or different kind of accident from any accident previously evaluated. The removal of the footnotes does not affect plant systems or operation. The footnotes were temporarily established to implement a conservative cycle specific MCPR Safety Limit until the SPC generic methodology was approved. With the approval of References 3 and 7, these footnotes are no longer applicable. Removing these footnotes does not introduce any physical changes to the plant, the processes used to operate the plant, or allowable modes of operation. The removal of the Unit 2 specific ''a'' pages, 2–1a and B2–3a, in the Quad Cities Technical Specifications, which is justified by the removal of the footnotes, also does not create a new or different kind of accident from any accident previously evaluated.

f. Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle 1 and 2)

The revision of the APLHGR and LHGR limit descriptions will not create the possibility of a new or different kind of accident from any accident previously evaluated. This revision will not alter any plant systems, equipment, or physical conditions of the site. This revision allows the flexibility of the APLHGR and the LHGR limits to be specified in the COLR and to maintain consistency with the calculated results of methodologies currently used to determine the APLHGR. The definition of the Average Planar Exposure is deleted, because it is being removed from LHGR and APLHGR Technical Specifications. This change does not introduce any physical changes to the plant, the processes used to operate the plant, or allowable modes of operation. Therefore this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in the margin of safety for the following reasons:

a. Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

The revised jet pump model methodology, and the MAPLHGRs, resulting from the revised jet pump methodology, will continue to ensure fuel design criteria and 10 CFR 50.46 compliance. The results of LOCA analyses performed with this methodology must continue to comply with the requirements of 10 CFR 50.46. Therefore, there is no significant reduction in the margin of safety.

b. Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

The margin of safety is not decreased by adding Reference 3 to Section 6.9.A.6.b of the Quad Cities Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications. Siemens Power Corporation methodology for application of the ANFB Critical Power Correlation to coresident GE fuel is approved by the NRC and is the same methodology used in the cycle specific topicals for coresident fuel (Reference 4 and 5). The MCPR Safety Limit will continue to ensure that greater than 99.9% of the rods in the core avoid boiling transition. Additionally, operating limits will be established to ensure the MCPR Safety Limit is not violated during all modes of operation.

c. Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The MCPR Safety Limit provides a margin of safety by ensuring that less than 0.1% of the rods are expected to be in boiling transition if the MCPR Safety Limit is not violated. This Technical Specification amendment request proposes to insert the topical report that describes SPC's calculation of the ATRIUM–9B additive constant uncertainty. The new ATRIUM–9B additive constant uncertainty calculation is conservative and is based on a larger database than previous calculations. Because the criteria of ensuring that 99.9% of the rods are expected to avoid boiling transition has not been changed and a conservative method is used to calculate the ATRIUM–9B additive constant uncertainty, a decrease in the margin to safety will not occur due to adding this methodology to the Technical Specifications. In addition, operational limits will be established to ensure the MCPR Safety Limit is protected for all modes of operation. This revised methodology will ensure that the appropriate level of fuel protection is being employed.

d. Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2, Dresden Unit 3, and LaSalle Units 1 and 2)

Changing the MCPR Safety Limit for Quad Cities Units 1 and 2, Dresden Unit 3, and LaSalle Units 1 and 2 will not involve any reduction in margin of safety. The MCPR Safety Limit provides a margin of safety by ensuring that less than 0.1% of the rods are calculated to be in boiling transition if the MCPR Safety Limit is not violated. The proposed Technical Specification amendment request reflects the MCPR Safety Limit results from conservative evaluations by SPC using the ANFB critical power correlation with the ATRIUM–9B additive constant uncertainty resulting from approval of Reference 7 (Reference 20).

Because a conservative method is used to apply the ATRIUM–9B additive constant uncertainty in the MCPR Safety Limit calculation, a decrease in the margin to safety will not occur due to changing the MCPR Safety Limit. The revised MCPR Safety Limit will ensure the appropriate level of fuel protection. Additionally, operational limits will be established based on the proposed MCPR Safety Limit to ensure that the MCPR Safety Limit is not violated during all modes of operation including anticipated operation occurrences. This will ensure that the fuel design safety criterion of more than 99.9% of the fuel rods avoiding transition boiling during normal operation as well as during an anticipated operational occurrence is met.

e. Removal of Footnotes Limiting Operation With ATRIUM–9B Fuel Reloads (Quad Cities Unit 2 and Dresden Units 2 and 3)

The removal of the cycle specific footnotes in Quad Cities and Dresden Technical Specifications does not impose a change in the margin of safety. These footnotes were added due to concerns regarding the calculation of the additive constant uncertainty for the ATRIUM–9B fuel and the cycle specific application of the ANFB critical power correlation to coresident GE fuel in Quad Cities Unit 2 Cycle 15. Because the generic ANFB application to coresident GE fuel MCPR methodology (Reference 3) has received NRC approval and the topical report describing the increased database used to calculate the additive constant uncertainties for ATRIUM–9B (Reference 7) has also received NRC approval (Reference 20) and both are proposed to be added to the Technical Specifications in this amendment request, there is no reason for the footnotes to remain. Removal of the Unit 2 specific ''a'' pages, 2–1a and B2–3a, in the Quad Cities Technical Specifications is justified by the removal of the footnotes. Therefore, the removal of the ''a'' pages, 2–1a and B2–3a, also does not impose a change in the margin of safety.

f. Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The revision to the APLHGR and LHGR limit descriptions will not involve a reduction in the margin of safety. The methodology used to calculate the APLHGR must comply with the guidelines of Appendix K of 10 CFR Part 50, and the APLHGR and LHGR will still be required to be maintained within the limits specified in the COLR. The surveillance requirements for these two thermal limits remain unchanged. Thus, there will be 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: For Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, IL 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, IL 61021; and for LaSalle, the Jacobs Memorial Library, 815 North Orlando Smith Avenue, Illinois Valley Community College, Oglesby, IL 61348–9692.

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

NRC Project Director: Stuart A. Richards.

Entergy Operations Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: July 17, 1996.

Description of amendment request: The proposed change extends the surveillance interval for the Reactor Trip Breakers (RTBs) from monthly to quarterly and increases the allowed outage time for operation with an inoperable RTB from one hour to two hours.

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. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

Response: No.

The proposed change to increase RTB surveillance interval will have no significant effect on the probability or consequences of any accident previously evaluated. As previously stated, all of the transient and accident analyses that call for a reactor trip assume that the reactor trip breakers (RTBs) operate and interrupt power to the control element drive mechanism (CEDMs). Extensive testing results, indicate that the RTBs are available and capable of performing their safety-related function. Currently RTBs are verified operable every 4 weeks. Under the proposed change RTBs would be verified operable at least every 6 weeks. This reduced testing frequency is intended to increase component reliability. The increase in the testing interval cannot increase component failure rate or the potential for component failure.

The proposed change to increase the allowed outage time for RTBs from 1 hour to 2 hours will have no significant impact on probability or consequences of any accident previously evaluated. When an RTB is inoperable, Functional Testing and other breaker operations becomes more difficult. The current technical specification allows an inoperable breaker to be closed for 1 hour to perform testing of other RTBs. This provision is infrequently required, but when it is required, the allowed outage time is very short and rushing to complete a test may lead to an inadvertent reactor trip. Increasing this allowed outage time is an improvement item identified in NUREG 1366 and consistent with philosophy provided in Generic Letter 89–07.

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

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different type of accident from any accident previously evaluated?

Response: No.

This proposed change does not involve any changes in equipment and will not alter the manner in which the plant will be operated.

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

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change will not adversely affect the performance of the safety function of the RTBs. In fact, it is expected that the performance of the RTBs will improve as a result of this change based on less wear and tear on the equipment. The proposed change will have no adverse impact on the protective boundaries, safety limits or margin of safety.

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

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

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

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

NRC Project Director: John N. Hannon.

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50–277, Peach Bottom Atomic Power Station, Unit No. 2, York County, Pennsylvania

Date of application for amendment: July 10, 1998.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to incorporate revised Safety Limit Minimum Critical Power Ratios (SLMCPRs) for the use of cycle-specific analysis performed for Peach Bottom Atomic Power Station (PBAPS), Unit 2, Cycle 13.

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

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

The derivation of the cycle-specific SLMCPRs for incorporation into the TS, and its use to determine cycle-specific thermal limits, have been performed using the methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–13, and U.S. Supplement, NEDE–24011–P–A–13–US, August, 1996, and the ''Proposed Amendment 25 to GE Licensing Topical Report NEDE–24011–P–A (GESTAR II) on Cycle Specific Safety Limit MCPR.'' Amendment 25 was submitted by [General Electric Nuclear Energy] GENE to the U.S. Nuclear Regulatory Commission (USNRC) on December 13, 1996. This change in SLMCPRs cannot increase the probability or severity of an accident.

The basis of the SLMCPR calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling and fuel damage in the event of a postulated accident. The fuel licensing acceptance criteria for the SLMCPR calculation apply to PBAPS, Unit 2, Cycle 13 in the same manner as they have applied previously. The probability of fuel damage is not increased. Therefore, the proposed TS changes do not involve an increase in the probability or consequences of an accident previously evaluated.

In addition to the change to the SLMCPR, the footnote to TS 2.1.1.2 is being revised, and a footnote is being added to TS 5.6.5.b.1. The revision to the footnote associated with TS 2.1.1.2 will ensure that the SLMCPR value is reconfirmed for the cycle subsequent to PBAPS, Unit 2, Cycle 13, and the footnote to TS 5.6.5.b.1 is being added due to the use of the proposed Amendment 25 and the use of a proposed R-factor calculation methodology (''R-Factor Calculation Method for GE11, GE12, and GE13 Fuel,'' NEDC–32505P, Revision 1, June 1997), which has not yet been approved for generic use by the USNRC. The revision to the footnote associated with TS 2.1.1.2 and the addition of the footnote to TS 5.6.5.b.1 are administrative changes that do not involve an increase in the probability or consequences of an accident previously evaluated.

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

The SLMCPR is a TS numerical value, designed to ensure that transition boiling does not occur in 99.9% of all fuel rods in the core during the limiting postulated accident. It cannot create the possibility of any new type of accident. The new SLMCPRs are calculated using methodology discussed in ''Generic Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–13, and U.S. Supplement, NEDE–24011–P–A–13–US, August, 1996, and the ''Proposed Amendment 25 to GE Licensing Topical Report NEDE–24011–P–A (GESTAR II) on Cycle Specific Safety Limit MCPR.'' Amendment 25 was submitted by GENE to the USNRC on December 13, 1996. Therefore, the revision to the SLMCPR will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Additionally, this proposed change will revise the footnote to TS 2.1.1.2, and add a footnote to TS 5.6.5.b.1. The revision to the footnote associated with TS 2.1.1.2, and the addition of the footnote to TS 5.6.5.b.1, are administrative changes that do not create the possibility of a new or different kind of accident from any previously evaluated.

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

There is no significant reduction in the margin of safety previously approved by the USNRC as a result of the proposed change to the SLMCPR, and the proposed change that will revise the footnote to TS 2.1.1.2, and add a footnote to TS 5.6.5.b.1. The new SLMCPRs are calculated using methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–13, and U.S. Supplement, NEDE–24011–P–A–13–US, August, 1996, and the ''Proposed Amendment 25 to GE Licensing Topical Report NEDE–24011–P–A (GESTAR II) on Cycle Specific Safety Limit MCPR.'' Amendment 25 was submitted by GENE to the USNRC on December 13, 1996. The fuel licensing acceptance criteria for the calculation of the SLMCPR apply to PBAPS, Unit 2 Cycle 13 in the same manner as they have applied previously. The SLMCPRs ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity. Therefore, the proposed TS changes will not significantly reduce the margin of safety previously approved by the USNRC.

Additionally, the proposed change that will revise the footnote to TS 2.1.1.2, and add a footnote to TS 5.6.5.b.1 is an administrative change that will not significantly reduce the margin of safety previously approved by the USNRC.

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, PA 17105.

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

NRC Project Director: Robert A. Capra.

Pennsylvania Power and Light Company, Docket No. 50–388 Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania.

Date of amendment request: August 4, 1998

Description of amendment request: The amendment would modify the Susquehanna Steam Electric Station, Unit 2, Technical Specifications to replace figures 2.1.1.2–1 and 2.1.1.2–2, and associated footnotes, with single value minimum critical power ratio (MCPR) Safety Limits of Section 2.1.1.2; remove references from Section 5.6.5 which do not directly support the generation of Core Operating Limits; remove references from Section 5.6.5 which were previously included to address the application of the ANFB–10 correlation to ATRIUM–10 fuel; include Siemens Power Corporation ANFB–10 topical report in Section 5.6.5; and to change the Bases to reflect inclusion of the ANFB–10 critical power correlation.

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

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

The applicable sections of the FSAR [Final Safety Analysis Report] are Chapters 4.4 and 15. FSAR Chapter 4.4 describes the MCPR Safety Limit, and Chapter 15 describes the transient and accident analyses. The reference to be added to Section 5.6.5 of the Unit 2 Technical Specifications describes an NRC approved critical power correlation for ATRIUM™ – 10 fuel appropriate for use in conservative methodologies for generating MCPR Safety Limits and MCPR Operating Limits to assure safe operation of Unit 2 with ATRIUM™ – 10 fuel. A discussion of the impact of the proposed Technical Specification change is provided below.

The proposed change in critical power correlation does not physically affect the plant or its systems. Thus, it does not increase the probability of an accident previously evaluated.

A Unit 2 Cycle 10 MCPR Safety Limit analysis was performed for PP&L by SPC. This analysis used NRC approved methods described in ANF–524(P)(A), Revision 2 and Supplement 1 Revision 2. These methods will be used each cycle to calculate the Unit 2 Safety Limits. For Unit 2 Cycle 10, the critical power performance of the 9x9 – 2 and ATRIUM™ – 10 fuel was determined using the NRC approved ANFB and ANFB – 10 correlations, respectively. The SAFETY LIMIT MCPR calculations statistically combine uncertainties on feedwater flow, feedwater temperature, core flow, core pressure, core power distribution, and uncertainties in the Critical Power Correlations. The SPC analysis used cycle specific power distributions and calculated MCPR values such that at least 99.91% of the fuel rods are expected to avoid boiling transition during normal operation or anticipated operational occurrences. The resulting two-loop and single-loop MCPR Safety Limits are included in the proposed Technical Specification change. Thus, the cladding integrity and its ability to contain fission products are not adversely affected.

Analyses of the Single Loop Pump Seizure accident with the NRC approved ANFB – 10 correlation for the ATRIUM™ – 10 fuel (Reference 1) [Reference 1 refers to the reference listed in the application dated August 4, 1998] will be performed to demonstrate that the NRC acceptance criterion (i.e., small fraction of 10 CFR 100 dose limits) is met. Analyses will also be performed to validate the conclusion that single-loop transients are less severe than the those events analyzed for two-loop operation.

Changes to Section 2.1.1.2 reflect the change from a flow dependent MCPR Safety Limit to a single value MCPR Safety Limit for two-loop operation and single-loop operation.

Changes to Reference 5.6.5 delete the methodology used for critical power analyses for ATRIUM™ – 10 fuel and add the NRC approved ANFB – 10 methodology to the list of approved methodologies. Other changes in Reference 5.6.5 are administrative in nature because they delete references that are not directly related to the generation of Core Operating Limits. No new analysis approaches are used due to the removal of these references.

Changes to BASES Sections 2.1.1 and 3.2.2 reflect the inclusion of the ANFB–10 critical power correlation. The range of the applicability of the ANFB–10 is valid for pressures > 571 psia and bundle mass fluxes > 0.115×10^6 lb/hr-ft^2. These values assure that a valid CPR calculation will result at or above 25% of rated core thermal power, that is, reactor steam dome pressure ≥785 psig and core flow ≥10 Mlbm/hr.

The consequences of transients and accidents will remain within the criteria approved by the NRC. The methodology used to perform the analyses have been previously approved by the NRC. Thus, analysis results using the new methodology will continue to provide assurance that the reactor will perform its design safety function during normal operation and design basis events. Therefore, the proposed action does not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes to the Unit 2 Technical Specifications (MCPR Safety Limits, removal of methodology references not directly supporting the generation of Core Operating Limits, removal of the two references describing previously approved methodology for applying ANFB to ATRIUM™–10 fuel, and inclusion of the ANFB–10 correlation reference) do not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. Removal of the Unit 2 Cycle 9 footnote allows Unit 2 Cycle 10 and future cycle operation with thermal limits generated using NRC approved methodology. Thus, the proposed change does not create the possibility of a previously unevaluated operator error or a new single failure. The consequences of transients and accidents will remain within the criteria approved by the NRC. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The applicable Technical Specification Sections include 2.1.1.2 and 5.6.5.

The changes to the Unit 2 Technical Specifications discussed in item 1 above do not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. Therefore, the proposed change will not jeopardize or degrade the function or operation of any plant system or component governed by Technical Specifications. The consequences of transients and accidents will remain within the criteria approved by the NRC. The proposed MCPR Safety Limits and use of the NRC approved ANFB–10 critical power correlation described in the reference added to Section 5.6.5 do not involve a significant reduction in the margin of safety as currently defined in the BASES of the applicable Technical Specification sections.

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

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

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

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

NRC Project Director: Robert A. Capra.

Pennsylvania Power and Light Company (PP&L), Docket No. 50–388, Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania

Date of amendment request: August 5, 1998.

Description of amendment request: The amendment would modify the Susquehanna Steam Electric Station, Unit 2, Technical Specifications Table 3.3.5.1–1 ''Emergency Core Cooling System Instrumentation.'' The change updates the allowable values for both the Core Spray (CS) and Low Pressure Coolant Injection System (LPCI) ''Reactor Steam Dome Pressure—Low'' functions for initiation and injection permissive. Specifically, the allowable values are changed from a specified minimum pressure to a specified allowable pressure band. This more restrictive allowable value range will prevent CS and LPCI system overpressurization while still permitting injection to prevent fuel clad temperature limits from being exceeded.

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

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

This proposal does not involve an increase in the probability or consequences of an accident previously evaluated. The proposed amendment changes the ''Reactor Steam Dome Pressure—Low'' Allowable Values so to provide further assurance that the Core Spray and [Residual Heat Removal] RHR systems will perform their [loss-of-coolant accident] LOCA design basis function.

The functional design basis of the Core Spray and LPCI is to inject water into the reactor vessel to cool the core during a LOCA by opening the Core Spray and LPCI injection valves when reactor pressure drops below the reactor vessel low pressure permissive. The upper analytical limit for the permissive is the Core Spray and LPCI systems' maximum design pressure, and the lower analytical limit is the lowest pressure which allows injection to prevent exceeding the fuel cladding temperature limit. The new allowable values were selected to lie within the upper and lower limits to ensure there will be no change in the required logic or functions of the Core Spray and LPCI systems. These new values do not affect the LOCA nor its ''limiting fault'' frequency of occurrence and do not introduce any new accidents or malfunctions of equipment important to safety. Since they do not affect the LOCA, they do not change the probability of occurrence of the LOCA. The new allowable values do not change the logic or function of the reactor vessel low pressure permissive. These new values simply provide the basis for which the associated pressure instruments are to be set to ensure proper operation of Core Spray and LPCI within the design pressures as described above. Therefore, the change in allowable values does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety.

Based upon the analysis presented above, PP&L concludes that the proposed action does not involve an increase in the probability or consequences of an accident previously evaluated.

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

This proposal does not create the probability of a new or different type of accident from any accident previously evaluated. The new allowable values do not change any plant systems, structures, or components, nor do they change any existing or create any new Core Spray and LPCI logic or functions. The new allowable values were selected to ensure the required operation of the Core Spray and LPCI systems within the maximum design pressures.

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

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

The change does not involve a reduction in the margin of safety. Technical Specification Bases Section B3.3.5.1 [9 sic] (ECCS Instrumentation) identifies that the low reactor steam dome pressure signals are used as permissives for operation of the low pressure ECCS subsystems. The new allowable values were selected so as to not impact the logic, redundancy, operability or surveillance requirements for these subsystems. The new allowable values maintain the margin requirements of the Core Spray and LPCI system pressures such that they do not exceed their system maximum design pressures and that system pressures are high enough to ensure that the ECCS injection prevents the fuel peak cladding temperature from exceeding the limits of 10 CFR50.46.

Therefore, the margin of safety is enhanced by the proposed changes.

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

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

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

NRC Project Director: Robert A. Capra.

Power Authority of the State of New York, Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: August 3, 1998.

Description of amendment request: The proposed changes provide for applicability of the safety limit minimum critical power ratio (SLMCPR) to fuel cycle 14.

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:

Operation of the FitzPatrick plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

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

A change to a note stating that the SLMCPR remains applicable through Cycle 14 does not affect the initiation of any accident. Operation in accordance with the current SLMCPR ensures the consequences of previously analyzed accidents are not changed. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The SLMCPR establishes a performance limit for the fuel. This limit remains unchanged. Changing a note to reflect this is an administrative change and will not initiate any accident. Therefore, this proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

GE [General Electric] has performed an evaluation of the SLMCPR for Cycle 14 and found that the cycle specific value, based on current reload plans, is bounded by the generic value calculated for GE 12 fuel. The existing SLMCPR remains unchanged for Cycle 14 and the margin of safety for the prevention of onset of transition boiling is unchanged. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Project Director: S. Singh Bajwa, Director.

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

Date of amendment request: July 30, 1998.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3/4.7.6, ''Control Room Emergency Air Conditioning System.'' Specifically, the acceptance criteria for the control room envelope would be revised to maintain a ⅛-inch positive pressure with respect to all areas directly accessible from the control room and a positive pressure with respect to all other areas adjacent to the control room.

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

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

CREACS [Control Room Emergency Air Conditioning System] ensures adequate protection after an accident and is not an accident initiator. The change to the acceptance criteria for CREACS does not affect the probability of an accident.

Revising the acceptance criteria for the CREACS from a '1/8-inch W.G. [water gauge] positive pressure in the control room with respect to the adjacent area' to 'a 1/8-inch W.G. positive pressure in the control room with respect to all areas directly accessible (Work Control Center and Control Room Equipment Rooms) from the control room and a positive pressure to all other areas adjacent to the control room' does not alter the assumptions in the radiological dose assessment provided to the NRC and approved under Amendments 190 (Unit 1) and 173 (Unit 2). Therefore the conclusions of the radiological dose assessment reviewed and approved by the NRC under the above Amendments remain unchanged. The radiological dose assessment provided under Amendments 190 and 173 demonstrates that operation of the CREAS in the pressurized mode at the initiation of an accident will ensure that the requirements of General Design Criterion (GDC) 19 will be met.

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

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

Since the CREACS is an accident mitigation system that does not communicate with the Reactor Coolant Pressure boundary or interface with Emergency Core Cooling Systems (ECCS), the proposed change to the acceptance criteria for CREACS pressurization cannot result in new accident scenarios. The function of the CREACS system is to maintain the habitability of the CRE [control room envelope] following an accident.

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

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

The CREACS ensures that (1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for equipment and instrumentation cooled by the CREACS and (2) the Control Room will remain habitable for operations personnel during and following all credible radiological accident conditions. Revising the acceptance criteria to maintaining the control room at a ⅛-inch W.G. positive pressure in the control room with respect to all areas directly accessible (Work Control Center and Control Room Equipment Rooms) from the control room and a positive pressure to all other areas adjacent to the control room does not alter the assumptions used in the radiological dose assessment nor revise the conclusions of the dose assessment which was reviewed under Amendments 190 and 173. Since the assumptions and conclusions of the dose assessment remain unchanged, the CREACS continues to ensure that the requirements of GDC 19 continue to be met, and there is no reduction in the safety provided to the control room operators.

Therefore, the proposed change to the TS does not involve a reduction in the margin of safety.

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

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

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: August 12, 1998.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3/4.6.1.3, ''Containment Air Locks,'' to change the action statements for an inoperable airlock. The proposed amendments would also correct an editorial error in TS Bases 3/4.6.1.2, ''Containment Leakage.''

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

The reactor containment serves to mitigate the consequences of a Design Basis Accident (DBA). That is, the containment is designed to provide a barrier to ensure that in the event of a DBA, a release of radioactive material will not result in the radiation dose to the general public exceeding the limits of 10 CFR 100. Each unit's containment has been provided with two air locks. These air locks permit personnel to access components and systems within the containment boundary without compromising the containment's ability to carry out its design function. In this capacity, the air locks serve as part of the containment boundary and as such are not considered as a contributor to the probability of an accident.

To carry out their design function, the air locks are designed and tested to certify their ability to withstand a pressure in excess of the maximum expected following a DBA. Each door is individually tested to verify that leakage will remain below design values with the containment at design pressure. An interlock is provided to ensure that containment integrity is maintained during personnel passage by allowing only one air lock door to be open at a time. This interlock is also periodically tested to verify its functionality.

The proposed changes will allow continued operation with one air lock door inoperable or with the air lock door interlock mechanism disabled but will specify the actions necessary under those conditions to assure that containment integrity is not compromised. This will ensure that the consequences of an accident previously evaluated are not significantly increased. Additionally, the proposed changes specify that in the event that an air lock is inoperable for a reason other than an inoperable air lock door, or air lock interlock mechanism, the unit must be placed in a condition in which the analyzed accident could not occur.

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

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

The proposed changes to the Containment Air Lock Technical Specifications do not affect the ability of the containment to carry out its design function. The changes also do not introduce any new equipment; nor do they result in the operation of the plant in a manner contrary to the safety analysis. Therefore, the proposed changes will not increase the probability of a new or different kind of accident from any accident previously identified.

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

The proposed changes do not affect any design or functional requirements of the Containment or the Containment Air Locks. Additionally, the proposed changes do not affect any of the conditions or assumptions of the applicable safety analyses. Containment Air Lock leakage rates are determined based upon containment leakage at design pressure. The proposed changes will not affect containment design pressure nor will they affect the peak containment pressures expected for analyzed accidents.

Based upon the above, the proposed change will not involve a reduction in a margin of safety.

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

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

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

NRC Project Director: Robert A. Capra.

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

Date of amendment requests: May 11, 1998 (Supersedes the May 30, 1996, amendment request). This Notice supersedes the staff's proposed no significant hazards consideration determination for the requested changes that was published on September 11, 1996 (61 FR 47981).

Description of amendment requests: The proposed amendments would revise the Technical Specifications (TS) to allow use of performance-based criteria to establish containment leak rate test intervals and add a new ''Containment Leakage Rate Testing Program'' to the administrative section of TS to codify the program used to determine the testing program. The proposed program implements 10 CFR Part 50, Appendix J, Option B, by referring to Regulatory Guide 1.163, ''Performance-Based Containment Leak Test Program,'' dated September 1995.

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

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

Since the interval between containment leakage rate tests is not related in any way to conditions which cause accidents, and plant structures, systems, and components will not be operated in a different manner as a result of the proposed Technical Specification (TS) change, the proposed changes will not increase the probability of an accident previously evaluated.

Containment leakage may result from accidents which are evaluated in the Updated Final Safety Analysis Report. The proposed TS changes may result in an acceptably small increase in post-accident containment leakage. Using a statistical approach, NUREG–1493 determined that the increase in hypothetical dose to the public resulting from extending the testing interval is extremely small. NUREG–1493 concluded that such small hypothetical dose increases to the public are justifiable due to the real reduction in occupational exposure resulting from interval extension. Therefore, the proposed change does not significantly increase the consequences of an accident previously evaluated.

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

The proposed change only incorporates the performance based approach for containment leak rate testing authorized in the new Option B to Appendix J of 10 CFR Part 50. The interval extensions allowed, through this approach, do not have the potential for creating the possibility of new or different kinds of accidents from those previously evaluated because plant structures, systems, and components will not be operated in a different manner as a result of the TS change and, therefore, will not introduce any new or different failure modes or initiators. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed Technical Specification does not alter the allowable containment leakage rate. The proposed change replaces the current, prescriptive testing requirements with a new performance based approach for establishing the testing intervals. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Main Library, University of California, Irvine, California 92713.

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

NRC Project Director: William H. Bateman.

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

Date of amendment requests: June 19, 1998.

Description of amendment requests: The proposed amendments would revise Technical Specification (TS) 3.4.1, ''RCS DNB (Pressure, Temperature and Flow) Limits.'' Specifically, the proposed changes would include (1) a reduction in the minimum primary reactor coolant system (RCS) cold leg temperature (T_{cold}) from 554 F to 535 F between the 70 percent and 100 percent rated thermal power levels, (2) a conversion of the specified RCS minimum flow rate from a ''Mass'' (i.e., lb/hr) to a ''Volumetric'' (gpm) flow basis, and (3) elimination of the maximum RCS flow rate limit.

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

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

The proposed change to Technical Specification (TS) 3.4.1 does not adversely impact structure, system, or component design or operation in a manner which would result in a change in the frequency of occurrence of accident initiation. Nor are the affected parameters themselves accident initiators. As such, the proposed TS change will not significantly increase the probability of accidents previously evaluated. Likewise, the proposed TS change does not significantly increase the consequences of an accident previously evaluated. The safety analysis assessments confirm that the existing Analyses of Record (AORs) for San Onofre Units 2 and 3 remain valid or have been re-analyzed to demonstrate continued compliance with applicable Acceptance Criteria.

The change in Reactor Coolant System (RCS) ''Mass'' flow to ''Volumetric'' flow is a change in measuring units to be consistent with the measure used in the performance of the safety analysis. Therefore, there is no impact on any evaluated accidents.

The elimination of the upper RCS flow limit has no effect on Departure from Nucleate Boiling which is a concern at lower flows, and the maximum flow that is physically possible is less than the current upper limit.

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

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

T_{cold} is an input parameter used in event analysis, it is not an event initiator. No new or different accidents have been identified which could result from operating at the proposed T_{cold}. The safety analysis assessments performed confirm that the existing safety system settings for San Onofre Units 2 and 3 remain valid, thereby assuring continued conformance to the Acceptance Criteria for all events.

A change in RCS flow measuring units can not initiate an accident, nor can the elimination of an upper RCS flow limit which can not be attained.

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

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

Updated Final Safety Analysis Report (UFSAR) safety analyses have been assessed and remain valid or have been re-analyzed to demonstrate continued compliance with applicable Acceptance Criteria for operation at the reduced T_{cold}. All other safety limits and safety system settings remain unchanged.

A change in measuring units for RCS flow does not reduce the margin of safety.

Elimination of an RCS flow limit that can not physically be reached does not reduce the margin of safety. The shiftly surveillance requirement for maximum flow has no practical basis or safety benefit. Additionally, the margin to departure from nuclear boiling increases as the flow rate increases.

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

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

Local Public Document Room location: Main Library, University of California, Irvine, California 92713.

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

NRC Project Director: William H. Bateman.

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

Date of amendment request: July 6, 1998.

Description of amendment request: The proposed amendment would relocate the description of the reactor coolant system design features from Technical Specification 5.4 to the Updated Final Safety Analysis Report.

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

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

The proposed change relocates the description of the Reactor Coolant System design features to the Updated Final Safety Analysis Report (UFSAR), a licensee-controlled document. The description of the Reactor Coolant System design features, currently a part of the UFSAR, is maintained in accordance with 10 CFR 50.59 and 50.71. Existing South Texas Project procedures ensure that changes to the facility as described in the UFSAR, such as the replacement of the steam generators, are reviewed to determine if an unreviewed safety question exists. The proposed amendment does not result in any hardware or operating procedure changes. The initiators of any accident previously evaluated are not affected by the relocation of the Reactor Coolant System design features. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The change does not alter the plant configuration or make changes in the methods governing plant operation. The proposed change does not impose different requirements, and adequate control of information will be maintained in accordance with existing procedures. The change does not alter assumptions made in the safety analysis and licensing basis. Therefore, the proposed change does not create the possibility of a new or different kind of accident.

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

The relocation of a description of Reactor Coolant System design features has no impact on any safety analysis assumptions. There are no changes to the plant configuration or operating procedures. Future changes to the relocated information are governed by existing procedures in accordance with 10 CFR 50.59 and 50.71. Consequently, there is no significant reduction in a margin of safety.

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

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

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

NRC Project Director: John N. Hannon.

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

Date of amendment request: July 6, 1998.

Description of amendment request: Relocates the Technical Specification 3/4.3.3.3 requirements for the Seismic Instrumentation to the Technical Requirements Manual.

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

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

The proposed change relocates requirements and surveillances for the Seismic Monitoring System that do not meet the criteria for inclusion in Technical Specifications as identified in 10 CFR 50.36(c)(2)(ii). The affected systems and components are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected systems and components will be relocated from the Technical Specifications to the Technical Requirements Manual, which is incorporated in the STP UFSAR and will be maintained pursuant to 10 CFR 50.59. In addition, the Seismic Monitoring System components are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. The associated changes to the Index are administrative. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change relocates requirements and surveillances for the Seismic Monitoring System that do not meet the criteria for inclusion in Technical Specifications as identified in 10 CFR 50.36(c)(2)(ii). The change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change will not impose different requirements, and adequate control of information will be maintained. This change will not alter assumptions made in the safety analysis and licensing basis. The associated changes to the Index are administrative. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change relocates requirements and surveillances for the Seismic Monitoring System, which does not meet the 10 CFR 50.36 criteria for inclusion in Technical Specifications. The change will not reduce a margin of safety because the change has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structures, systems, components, or variables remain the same as the existing Technical Specifications. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, there will be no reduction in a margin of safety. The associated changes to the Index are administrative and have no potential effect on the margin of safety.

Therefore, the 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 standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

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

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

NRC Project Director: John N. Hannon.

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

Date of amendment request: July 6, 1998.

Description of amendment request: Relocates the Technical Specification 3/4.7.13 requirements for the Area Temperature Monitoring System to the Technical Requirements Manual.

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

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

The proposed change relocates requirements and surveillances for Technical Specification 3/4.7.13, which does not meet the criteria for inclusion in Technical Specifications as identified in 10 CFR 50.36(c)(2)(ii). The affected systems and components are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected systems and components will be relocated from the Technical Specifications to the Technical Requirements Manual, which is incorporated in the STP UFSAR and will be maintained pursuant to 10 CFR 50.59. In addition, the Area Temperature Monitoring System components are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. The associated changes to the Index are administrative. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change relocates requirements and surveillances for the Area Temperature Monitoring System, which does not meet the criteria for inclusion in Technical Specifications as identified in 10 CFR 50.36(c)(2)(ii). The change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change will not impose different requirements, and adequate control of information will be maintained. This change will not alter assumptions made in the safety analysis and licensing basis. The associated changes to the Index are administrative. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change relocates requirements and surveillances for the Area Temperature Monitoring System, which does not meet the 10 CFR 50.36 criteria for inclusion in Technical Specifications. The change will not reduce a margin of safety since it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component, or variable remain the same as the existing Technical Specifications. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, there will be no reduction in a margin of safety. The associated changes to the Index are administrative and have no potential effect on the margin of safety.

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

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

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

NRC Project Director: John N. Hannon.

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

Date of amendment request: July 22, 1998.

Description of amendment request: The proposed amendment would revise the Technical Specifications to reflect the steam generator water level low-low trip setpoint differences between the existing Model E and the replacement Model Delta-94 steam generators for the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation.

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

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

This proposed change includes changing the low-low steam generator water level trip setpoint. The setpoint is being changed to enhance the operational flexibility associated with the RSGs [replacement steam generators].

The minimum setpoint change proposed in this request establishes controls to ensure that an adequate heat sink is maintained by providing an adequate secondary liquid mass to remove primary system sensible heat and core decay heat shortly after reactor trip and initiating auxiliary feedwater flow for long-term cooling. The accidents analyzed for this requirement are the Loss of Non-Emergency AC Power to the Plant Auxiliaries, Loss of Normal Feedwater and Feedwater Line Break transients. These accidents were analyzed utilizing the Westinghouse RETRAN model. All acceptance criteria were shown to be met for both these events. Therefore, the proposed steam generator water level low-low trip setpoint change is demonstrated not to result in an increase in the consequences for these accidents.

The steam generator water level low-low trip setpoint is not considered a precursor to any of the analyzed accidents, and therefore, these proposed changes do not result in an increase in the probability or consequences of any accident previously analyzed.

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

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

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

The events potentially affected by the setpoint change in the steam generator water level low-low reactor trip (Table 2.2–1, Function 13) and ESFAS Auxiliary Feedwater System actuation (Table 3.3–4, Function 6.d) are the Loss of Normal Feedwater and Feedwater System Pipe Break. These events were analyzed and it was demonstrated that all acceptance criteria were met for both of these events.

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

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

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

NRC Project Director: John N. Hannon.

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

Date of amendment request: July 28, 1998.

Description of amendment request: The proposed amendment addresses the operator action to reduce the steam generator power-operated relief valve setpoint consistent with the revised small-break loss-of-coolant accident (SBLOCA) analysis for the replacement Delta-94 steam generators. The operator action and the associated revised SBLOCA analysis are reflected in a proposed revision to the South Texas Project Updated Final Safety Analysis Report.

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

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

The proposed operator action associated with the re-analysis of the Delta-94 SGs [steam generators] will not result in a significant increase in the probability of an accident previously evaluated. The initiators of any design basis accident are not affected by this operator action. The operator action would facilitate the automatic mitigation capability of the SG PORVs [power-operated relief valves], and would not initiate the mitigating safety function. The operator action will be incorporated into the EOPs [Emergency Operating Procedures] and would not be performed until after the initiation of an accident. The automatic actuation of the SG PORVs is not a new design feature. The effects of inadvertent opening of a single steam dump, relief or safety valve are currently analyzed as described in Section 15.1.4 of the UFSAR [Updated Final Safety Analysis Report]. Consequently, there is no significant impact on any previously evaluated accident probabilities.

The proposed operator action associated with the re-analysis of the Delta-94 SGs does not result in a significant increase in the consequences of any accidents previously evaluated. The operator action will not adversely affect the integrated ability of the plant systems to perform their intended safety functions to mitigate the consequences of a small break LOCA [loss-of-coolant accident], or any other accident previously evaluated. In fact, the re-analysis has demonstrated that the use of the operator action reduces the consequences of a small break LOCA in that the Peak Cladding Temperature for the most limiting small break LOCA transient is reduced and continues to be substantially below the acceptance limit of 10 CFR 50.46.

The operator action does not affect the integrity of any fission product barrier such that their function in the control of radiological consequences is not affected. The radiological consequences for the small break LOCA presented in the UFSAR remain unchanged as a result of the proposed operator action.

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

The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment is not the result of any physical changes to the existing facility. The operator action does not represent a different initiator for any design basis accident and does not create new design basis scenarios. Small break LOCA mitigation, utilizing a combination of automatic and manual actions, is already part of the STP [South Texas Project, Units 1 and 2] licensing basis. Written procedures address those operator actions required for small break LOCA mitigation. The current STP EOPs have an operator action for a steam generator tube rupture (SGTR) similar to the operator action for the small break LOCA addressed by this proposed license amendment. The operator action for the SGTR is to raise the safety-grade SG PORV setpoints. The operator action credited in the small break LOCA analysis for the Delta-94 SGs is to lower the safety-grade SG PORV setpoints. The purpose of the action is to provide a more rapid cooldown of the primary side by depressurizing the secondary side during a small break LOCA using the steam dumps first, then the SG PORVs, if steam dumps are unavailable. The inadvertent operation of a single steam dump, relief or safety valve is currently addressed in UFSAR Section 15.1.4.

The proposed amendment does not alter any original design specification, such as seismic requirements, electrical separation requirements and environmental qualification, and is not the result of any physical changes to the facility. In addition, the proposed amendment does not result in exposure of additional equipment used in accident mitigation to an adverse environment beyond that currently identified in the UFSAR.

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

The proposed operator action does not involve a significant reduction in the margin of safety. The plant systems required for the mitigation of any design basis accidents will continue to be able to perform their safety function. In fact, the re-analysis has demonstrated that the use of the operator action reduces the consequences of a small break LOCA in that the Peak Cladding Temperature for the most limiting small break LOCA transient is reduced and continues to be substantially below the acceptance criteria of 10 CFR 50.46.

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

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

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

NRC Project Director: John N. Hannon.

Tennessee Valley Authority, Docket No. 50–259, 50–260 and 50–296, Browns Ferry Nuclear Plant Units 1, 2, 3, Limestone County, Alabama

Date of amendment request: June 12, and August 14, 1998.

Description of amendment request: The proposed amendment would revise the technical specifications (TS) for the Browns Ferry Nuclear Plant (BFN) Units 1, 2 and 3. The proposed changes would revise surveillance frequency of ''once-per-cycle'' surveillance requirements (SR) from 18 to 24 months to accommodate a 24-month fuel cycle. The licensee also proposed changes to the associated TS Bases (TS–390).

Basis for proposed no significant hazards consideration determination: Tennesee Valley Authority addressed the affected SRs into two groups: (1) non-instrument calibration related, and (b) those involving instrument calibrations. 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:

Group 1: Non-instrument Calibration Related SRs

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

The proposed amendment changes the surveillance frequency from 18 months to 24 months for SRs in the Units 2 and 3 TS that are normally a function of the refueling interval. In addition, the proposed amendment changes the surveillance frequency from 18 months to 24 months for those SRs in the Unit 1 TS that control the test interval for components and systems that are common to Units 1, 2, and 3. Under certain circumstances SR 3.0.2 would allow a maximum surveillance interval of 30 months for these SRs. The evaluations in Section III [Licensee's June 12, 1998 application, Section III, Safety Analysis] have shown that the reliability of protective instrumentation and equipment will be preserved for the maximum allowable surveillance interval. The proposed changes do not involve any change to the design or functional requirements of plant systems, and the surveillance test methods will be unchanged. The proposed changes will not give rise to any increase in operating power level, fuel operating limits, or effluents. In addition, the proposed changes will not significantly increase any radiation levels. Based on the foregoing considerations and the evaluations completed in accordance with the guidance of Generic Letter 91–04, it is concluded that the proposed amendment does not involve a significant 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.

The proposed amendment requires no change to the plant design or the mode of operation, for any item of equipment. No new equipment is either added or substituted for any existing equipment. Based on the Section III [Licensee's June 12, 1998 application, Section III, Safety Analysis] evaluations, the extension of surveillance intervals is shown to have no significant impact on equipment performance. The proposed changes do not create the possibility of any new failure mechanisms. Therefore, the proposed amendment does 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 a margin of safety.

The proposed amendment seeks to change surveillance intervals from 18 to 24 months. Although the proposed TS changes will result in an increase in the interval between surveillance tests, the impact on system availability is small based on other, more frequent testing or redundant systems or equipment. There is no evidence of any failures that would impact the availability of the systems. This change does not alter the existing setpoints, TS allowable values or analytical limits. The assumptions in the current safety analyses are not impacted and the proposed amendment does not reduce a margin of safety.

Therefore, it is concluded that the proposed amendment does not involve a significant reduction in a margin of safety.

Group 2: SRs that Involve Instrument Calibrations

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

The proposed amendment changes the surveillance Frequency from 18 months to 24 months for SRs in the Units 2 and 3 TS that are normally a function of the refueling interval. In addition, the proposed amendment changes the surveillance Frequency from 18 months to 24 months for those SRs in the Unit 1 TS that control the test interval for components and systems that are common to Units 1, 2, and 3. Under certain circumstances SR 3.0.2 would allow a maximum surveillance interval of 30 months for these SRs. The evaluations in Section III [Licensee's August 14, 1998 application, Section III, Safety Analysis] have shown that the reliability of protective instrumentation will be preserved for the maximum allowable surveillance interval. The proposed changes do not involve any change to the design or functional requirements of plant systems, and the surveillance test methods will be unchanged. The proposed changes will not give rise to any increase in operating power level, fuel operating limits, or effluents. In addition, the proposed changes will not significantly increase any radiation levels. Based on the foregoing considerations and the evaluations completed in accordance with the guidance of Generic Letter 91–04, it is concluded that the proposed amendment does not involve a significant 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.

The proposed amendment requires no change to the plant design or the mode of operation, for any item of equipment. The proposed changes do not create the possibility of any new failure mechanisms. Therefore, the proposed amendment does 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 a margin of safety.

The proposed amendment seeks to change instrument calibration surveillance intervals from 18 to 24 months. The primary consideration relative to safety margin is that of exceeding analytical limits for the current safety analyses as a result of increased instrument drift over the extended surveillance interval. The drift studies discussed in Section III.A have shown that the existing setpoints and TS allowable values can be retained without challenging the current analytical limits; thereby preserving the assumptions in the current safety analyses and ensuring that safety limits will not be exceeded.

To confirm that the drift errors remain within projected values, instruments subjected to the longer interval between calibrations will continue to be monitored as required by current plant procedures. This practice will assure that no significant reduction in safety margin is incurred by adoption of the proposed amendment.

Therefore, it is concluded that the proposed amendment does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on its 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: Athens Public Library, 405 E. South Street, Athens, Alabama 35611

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902.

NRC Project Director: Frederick J. Hebdon.

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

Date of application for amendments: August 22,1996 (TS 97–04), as supplemented on August 27, 1998.

Brief description of amendments: The amendments would change the Sequoyah (SQN) Technical Specifications (TS) by extending the emergency diesel generator allowed outage time from 72 hours to 7 days. This amendment request was previously noticed on October 9, 1996 (61 FR 52969). The scope of the amendment request was changed by the August 27, 1998 submittal.

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

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

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

The EDGs [emergency diesel generators] supply backup power to the essential safety systems in the event of a loss-of-offsite (normal) power. The EDGs are not postulated to be an initiator of a design basis accident. The requested change to provide a 7-day AOT [allowed outage time] for the EDGs and the deletion of the additional 72-hour extension for this AOT will not impact the plant design, components or operational practices. The increased out-of-service time does not invalidate assumptions used in evaluating the radiological consequences of an accident and does not provide a new or altered release path. In addition, the administrative changes to delete EDG reporting requirements and an obsolete License Condition will not impact plant equipment or operating practices. Therefore, this change does not involve an increase in the probability of any accident previously evaluated.

An increase in the AOT for the EDGs would not change the conditions, operating configuration, or minimum amount of operable equipment assumed in the plant Final Safety Analysis Report for accident mitigation. The longer AOT would provide a longer time window for maintenance, but would lesson the overall EDG unavailability, therefore, it would reduce plant risk. The CDF [core damage frequency] associated with a 7-day AOT increases from the base case in the SQN [Sequoyah Nuclear Plant] IPE [individual plant examination] but is not risk-significant. This CDF increase is based on sensitivity studies performed in accordance with the guidance in Draft Regulatory Guide DG–1065, dated June 1997. These studies assume additional unavailability of the EDGs for an increase in AOT even though plant practices are not expected to change. The EDG availability improvements and CDF reductions during 12- and 6-year maintenance activities compensates for this potential increase to provide an overall safety benefit.

The deletion of the footnote for extending the AOT for fuel tank cleaning removes inappropriate extensions of EDG out-of-service time. SQN's implementation of the Maintenance Rule, 10 CFR 50.65, also supports the proper scheduling and performance of maintenance activities to ensure EDG unavailability is adequately controlled. Based on no change in plant risk during routine maintenance, because work activity durations are unchanged, and the decrease in overall plant risk during the 12- and 6-year maintenance activities, as a result of the 7-day EDG action time, this change will not result in a significant increase in the consequences of an accident. In addition, the administrative deletions of reporting requirements that are not necessary based on Maintenance Rule implementation and obsolete License Condition deletion will not increase the consequences of an accident.

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

The proposed change to extend the AOT for the EDGs and delete unnecessary TS and operating license provisions does not alter the physical design or configuration of the plant. The EDG operation remains unchanged, therefore, this change does not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed extension of the EDG action time for inoperable units to 7 days will not alter plant equipment, setpoints or operating practices that provide the necessary margin of safety. The extension will reduce EDG unavailability and plant risk such that the

EDG's ability to react to accident situations is increased. Overall CDF, as a result of a 7-day AOT, indicates a slight increase but it is not significant. The AOT extension deletion for fuel tank cleaning is a conservative change to maintain appropriate EDG out-of-service times. The deletions of administrative requirements for reporting EDG reliability and obsolete License Conditions do not impact functions that maintain the margins of safety and have been or are continuing to be satisfied by other regulatory requirements. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Frederick J. Hebdon.

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

Date of application request: June 29, 1998.

Description of amendment request: The amendment would revise technical specification 3.7.1.7 to (1) address operability of all four atmospheric steam dump (ASD) lines, (2) retain an action statement for excessive ASD seat leakage, and (3) incorporate action statements for multiple inoperable ASD lines.

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

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

Revising the LCO to refer to the ASD lines rather than the ASD valves; requiring four ASD lines to be operable rather than three; limiting the LCO 3.0.4 exception to one ASD line inoperable; and adding a surveillance for the manual isolation valves constitutes a more restrictive change from the current Specification. The proposed changes impose more stringent requirements to ensure that ASD operability is maintained consistent with the safety analysis and licensing basis, and also to address all potential single failure scenarios.

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

When two ASD lines are inoperable due to causes other than excessive ASD seat leakage, the proposed change increases the allowed outage time for restoration of all but one required ASD line from 24 hours to 72 hours. The increase in time is not significant when balanced against the availability of the condenser steam dump system and/or the main steam safety valves, and the low probability of an event occurring during the restoration period that would require the ASD lines. Therefore the increase in allowed outage time for restoration of all but one ASD line does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change revising the required completion time from hot standby to hot shutdown from six hours to twelve hours is consistent with NUREG–1431, Rev. 1, where the required completion time to shut the plant down is revised to achieving hot standby in six hours and hot shutdown within the following twelve hours. The proposed change does not alter the plant configuration or operation or the function of any safety system. Consequently, the change does not increase the probability of an accident as defined in the accident analysis. The proposed change permits a longer time to cooldown to RHR entry conditions; however, this would not affect the consequences of any postulated accidents and is appropriate due to the need to avoid any transients while cooling down. Therefore the proposed change would not involve a significant increase in the probability or consequences of an accident.

Therefore, it is concluded that all of the above-proposed changes do not significantly increase the probability or consequences of any accident previously evaluated.

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

Revising the LCO to refer to the ASD lines rather than the ASD valves; requiring four ASD lines to be operable rather than three; limiting the LCO 3.0.4 exception to one ASD line inoperable; and adding a surveillance for the manual isolation valves does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in controlling parameters. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

When two ASD lines are inoperable due to causes other than excessive ASD seat leakage, the proposed change increases the allowed outage time for restoration of all but one required ASD line from 24 hours to 72 hours. The increase in time is not significant when balanced against the availability of the condenser steam dump system and/or the main steam safety valves, and the low probability of an event occurring during the restoration period that would require the ASD lines. The increase in the allowed outage time does not result in a condition not previously considered or analyzed, and therefore does not create the possibility of a new or different kind of accident.

The proposed change revising the required completion time from hot standby to hot shutdown from six hours to twelve hours is consistent with NUREG–1431, Rev. 1, where the required completion time to shut the plant down is revised to achieving hot standby in six hours and hot shutdown within the following twelve hours. The proposed change does not require physical alteration to any plant system or change the method by which any safety-related system performs its function. The change does allow additional time to complete the transfer from the steam generator method for heat removal to the RHR system, but does not alter the basic methodology. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

All of the proposed changes discussed above do not create the potential for a new or previously unanalyzed accident.

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

Revising the LCO to refer to the ASD lines rather than the ASD valves; requiring four ASD lines to be operable rather than three; limiting the LCO 3.0.4 exception to one ASD line inoperable; and adding a surveillance for the manual isolation valves imposes more stringent requirements. These requirements either have no impact on or increase the margin of safety by increasing the scope of the specification to include additional plant equipment; by adding additional requirements; and by imposing a new surveillance. The change is consistent with the safety analysis and licensing basis, and does not involve a reduction in a margin of safety.

When two ASD lines are inoperable due to causes other than excessive seat leakage, the proposed change increases the allowed outage time for restoration from 24 hours to 72 hours. The increase in time is not significant when balanced against the availability of the condenser steam dump system and/or the main steam safety valves, and the low probability of an event occurring during the restoration period that would require the ASD lines. The increase in the allowed outage time does not result in a condition not previously considered and does not involve a significant reduction in a margin of safety.

The proposed change revising the required completion time from hot standby to hot shutdown from six hours to twelve hours is consistent with NUREG–1431, Rev. 1, where the required completion time to shut the plant down is revised to achieving hot standby in six hours and hot shutdown within the following twelve hours. The change does not alter the basic regulatory requirements or change any accident analysis assumptions, initial conditions or results. Therefore, the proposed change would have no significant adverse effect on margins of safety.

None of the proposed changes have any significant adverse effect on margins 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: Elmer Ellis Library, University of Missouri, Columbia Missouri 65201.

Attorney for licensee: John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Project Director: William H. Bateman.

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

Date of amendment request: July 28, 1998.

Description of amendment request: The North Anna Power Station (NAPS), Unit 1 and 2, Technical Specifications (TS) Surveillance Requirement (SR) 4.6.2.2.1.b requires verification, during recirculation flow, that each outside recirculation spray (ORS) pump develops a discharge pressure of greater than or equal to 115 pounds per square inch (psig) and that each Casing Cooling pump develops a discharge pressure of greater than or equal to 58 psig for Unit 1 and 46 psig for Unit 2 when tested. The proposed changes will revise the testing acceptance criteria being verified from discharge pressure to the required developed head. The frequency of testing shall be in accordance with the Inservice Testing Program.

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 reviewed the requirements of 10 CFR 50.92 as they relate to the proposed changes for the North Anna Units 1 and 2 and determined that the changes do not pose a significant hazards consideration * * * Specifically, operation of the North Anna Power Station in accordance with the proposed Technical Specification changes will not:

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

The applicable UFSAR [Updated Final Safety Analysis Report] accidents previously evaluated are the LOCA [loss-of-coolant accident] and MSLB [main steamline break]. The proposed changes ensure that the Casing Cooling and ORS pumps will perform properly with no unacceptable degradation by using the correct pump test acceptance criteria as controlled by the PT program. This does not increase the probability of a LOCA or MSLB.

(b) Create the possibility of a new or different type from any accident previously evaluated

The proposed changes to the Technical Specifications will ensure that the Casing Cooling and ORS pumps are tested at the frequency established by the lnservice Testing Program to confirm their ability to provide design basis flow during a LOCA/MSLB. This will not result in any physical alteration to any plant system, nor would there be a change in the method by which any safety related system performs its function. The design and operation of the Casing Cooling and ORS systems are not being changed. Also, the proposed changes do not affect the design, operation or failure modes of the Casing Cooling and ORS pumps and other components within the Casing Cooling and ORS systems. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

(c) Involve a significant reduction in a margin of safety

Implementation of the proposed changes ensures that the Casing Cooling and ORS pumps do not operate with unacceptable degraded flows during a LOCA/MSLB that are less than their containment analysis design basis flow. Therefore, the proposed changes would not reduce the margin of safety.

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

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

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

NRC Project Director: Pao-Tsin Kuo, Acting.

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

Date of amendment request: July 28, 1998.

Description of amendment request: The North Anna Power Station (NAPS), Unit 1 and 2, Technical Specifications (TS) Surveillance Requirements (SR) 4.8.1.1.2.a.4, 4.8.1.1.2.c, 4.8.1.1.2.d.2, 4.8.1.1.2.d.4.b, 4.8.1.1.2.d.5, 4.8.1.1.2.d.6.b, 4.8.1.1.2.d.11.b, and 4.8.1.1.2.e currently require each Emergency Diesel Generator (EDG) to be demonstrated OPERABLE by the performance of specific Surveillance Requirements. One significant part of demonstrating operability of the EDG requires verification that the frequency is within a specified range, which is currently 60 plus or minus 1.2 Hz. The proposed changes would change the frequency limit from 60 plus or minus 1.2 Hz to 60 plus or minus 0.5 Hz and separate the requirement of the EDG start from the steady state voltage and frequency limits.

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 reviewed the proposed Technical Specification changes against the requirements of 10 CFR 50.92 and has determined that the proposed changes would not pose a significant hazards consideration. Specifically, operation of the North Anna Power Station in accordance with the proposed Technical Specifications changes will not:

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

The proposed change provides a more stringent requirement for the EDG frequency limit at steady state operation of 60 [plus or minus] 0.5 Hz from the current 60 [plus or minus] 1.2 Hz. The change additionally provides a separation of the start requirements from the steady state limits for voltage and frequency. The change to the EDG frequency limit does not result in operation that will increase the probability of initiating an analyzed event and does not alter assumptions relative to mitigation of an accident or transient event. The change to the frequency limit is acceptable because the safety analyses assumptions for emergency power limits the frequency variations to 60 [plus or minus] 0.5 Hz and assumes that the EDG supplies the emergency bus with electrical power within 10 seconds of receiving an emergency start signal. The EDG output breaker will close with no electrical power applied to the emergency bus when the EDG output reaches 95% of rated voltage. The minimum frequency requirement of 59.5 Hz is based on the steady state limit for the EDG. The EDG supplies the electrical power for the required equipment to mitigate the consequences of design basis events. The minimum voltage and frequency (3740 volts and 59.5 Hz) limits ensure that the ESF [engineered safety feature] equipment is maintained with the required electrical power to mitigate the consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change provides a more stringent requirement for the EDG frequency at steady state operation of 60 [plus or minus] 0.5 Hz from the current 60 [plus or minus] 1.2 Hz. The change additionally provides a separation of the start requirements from the steady state limits for voltage and frequency. The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change provides a more stringent requirement for the EDG frequency at steady state operation of 60 [plus or minus] 0.5 Hz from the current 60 [plus or minus] 1.2 Hz. The change additionally provides a separation of the start requirements from the steady state limits for voltage and frequency. The change to the frequency limit is acceptable because the safety analyses assumptions for emergency power limits the frequency variations to 60 [plus or minus] 0.5 Hz and assumes that the EDG supplies the emergency bus with electrical power within 10 seconds of receiving an emergency start signal. The EDG output breaker will close with no electrical power applied to the emergency bus when the EDG output reaches 95% of rated voltage. The minimum frequency requirement of 59.5 Hz is based on the steady state limit for the EDG.

The EDG supplies the electrical power for the required equipment to mitigate the consequences of design basis events. The minimum voltage and frequency (3740 volts and 59.5 Hz) limits ensure that the ESF equipment will be supplied with the required electrical power to mitigate previously evaluated accidents. The margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The change allowing the separation of the start requirements from the steady state voltage and frequency limits, due to the short time period allowed in this condition, does not significantly impact the performance of structures; systems or components relied upon for accident mitigation or any safety analysis assumptions. Therefore, the change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Pao-Tsin Kuo, Acting.

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

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

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

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

Date of amendment request: August 6, 1998.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) Surveillance Requirement 4.8.1.1.2.i.2. This requirement is in conflict with a relief granted by the NRC staff in February 1995. The deletion of TS Surveillance Requirement 4.8.1.1.2.i.2 would remove such a conflict.

Date of publication of individual notice in **Federal Register**: August 17, 1998 (63 FR 43962).

Expiration date of individual notice: September 16, 1998.

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

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

Date of application for amendment: July 16, 1998. This notice supersedes a previous notice (62 FR 40851, published July 30, 1997) that was based upon an amendment request dated July 2, 1997. The request dated July 2, 1997, was superseded in its entirety by the amendment request dated July 16, 1998.

Brief description of amendment: The amendment would change Technical Specification 3/4.2.3 regarding reactor coolant chemistry in accordance with a report by Electrical Power Research Institute, Inc. TR–103515–R1, ''BWR Water Chemistry Guidelines, 1996 Revision,'' also known as Boiling Water Reactor Vessel and Internals Project-29.

Date of publication of individual notice in **Federal Register**: August 13, 1998 (63 FR 43432).

Expiration date of individual notice: September 14, 1998.

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

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

Date of amendment requests: February 27, 1998, as supplemented July 14, 1998.

Brief description of amendment requests: The proposed amendments would allow a design modification to the existing Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC). The design modification would install a Diverse Scram System (DSS) designed to meet the requirements of a DSS described by 10 CFR 50.62 (ATWS Rule) for non-Westinghouse designed plants and make major modifications to the existing AMSAC.

Date of publication of individual notice in **Federal Register**: August 17, 1998 (63 FR 4365).

Expiration date of individual notice: September 16, 1998.

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

Power Authority of the State of New York, Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of application for amendment: October 14, 1997, as supplemented July 23, 1998.

Brief description of amendment: The proposed amendment would change the James A. FitzPatrick Technical Specifications to provide for installation of additional racks to increase spent fuel pool capacity, and to correct the maximum exposure dependent, infinite lattice multiplication factor for fuel bundles.

Date of initial notice in **Federal Register**: August 24, 1998 (63 FR 45096).

Expiration date of individual notice: September 23, 1998.

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

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.

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 14, 1998.

Brief description of amendments: The amendments revise the Technical Specifications to support replacement of the 125 volt direct current (Vdc) AT&T batteries with new Charter Power Systems, Inc. (C&D) batteries. In addition, the crosstie loading limitation is revised to reflect the larger capacity of the C&D batteries.

Date of issuance: August 18, 1998.

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

Amendment Nos.: 94 and 94.

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

Date of initial notice in **Federal Register**: May 20, 1998 (63 FR 27758). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 18, 1998.

No significant hazards consideration comments received: No.

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

Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan

Date of application for amendment: April 2, 1998 (NRC–98–0057).

Brief description of amendment: The amendment revises Technical Specification 3.3.7.5 to permit entering Operational Conditions 1 and 2 prior to completion of Surveillance Requirements for the primary containment hydrogen and oxygen monitors in order to establish the conditions necessary (inerted containment) to properly perform the calibrations. The amendment also allows an increase in the frequency of the calibration for the oxygen monitors from once every 18 months to quarterly and corrects the nomenclature for the hydrogen and oxygen monitors in tables 3.3.7.5–1 and 4.3.7.5–1.

Date of issuance: August 20, 1998.

Effective date: August 20, 1998, with full implementation within 90 days.

Amendment No.: 125.

Facility Operating License No. NPF–43. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: April 22, 1998 (63 FR 19968).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 20, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Monroe County Library System, Ellis Reference and Information Center, 3700 South Custer Road, Monroe, Michigan 48161.

Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan

Date of application for amendment: March 27, 1998 (NRC–98–0034), as supplemented May 28 and July 31, 1998.

Brief description of amendment: The amendment revises footnotes associated with the emergency core cooling system (ECCS) in Technical Specifications 3.5.1, ''ECCS—Operating,'' and 3.5.2, ''ECCS—Shutdown,'' to indicate that a low pressure coolant injection system loop may be considered operable during alignment and operation for decay heat removal if it is capable of being manually realigned and is not otherwise inoperable. The associated Bases are also revised.

Date of issuance: August 25, 1998.

Effective date: August 25, 1998, with full implementation within 90 days.

Amendment No.: 126.

Facility Operating License No. NPF–43: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: April 22, 1998 (63 FR 19968). The May 28 and July 31, 1998, letters provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards considerations determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 25, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Monroe County Library System, Ellis Reference and Information Center, 3700 South Custer Road, Monroe, Michigan 48161.

Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan

Date of application for amendment: September 25, 1996 (NRC–96–0085), as supplemented by letters dated November 26, 1997, and March 10 and June 17, 1998.

Brief description of amendment: The amendment revises Surveillance Requirement 4.8.4.3 to clarify the situational testing requirement for thermal overload devices to indicate that this portion of the requirement must be completed upon initial installation of a thermal overload device and following any maintenance that could affect its performance.

NRC has also granted the request of Detroit Edison Company to withdraw a portion of its September 25, 1996, application. The proposed change would have deleted the requirement for periodically testing motor-operated valve thermal overload protective devices. However, by letter dated June 17, 1998, the licensee withdrew this portion of the amendment request. For further details with respect to these actions, see the application for amendment dated September 25, 1996, as supplemented above, and the licensee's letter dated June 17, 1998, which withdrew this portion of the application for license amendment, and the staff's safety evaluation enclosed with the amendment. 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 listed below.

Date of issuance: August 25, 1998.

Effective date: August 25, 1998, with full implementation within 90 days.

Amendment No.: 127.

Facility Operating License No. NPF–43: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: October 23, 1996 (61 FR 55030).

The November 26, 1997, and March 10 and June 17, 1998, submittals provided additional clarifying information within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 25, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Monroe County Library System, Ellis Reference and Information Center, 3700 South Custer Road, Monroe, Michigan 48161.

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

Date of application for amendments: April 8, 1998.

Brief description of amendments: The amendments revise Technical Specification Section 3/4.6.5.1, regarding the ice condenser, to reduce the total ice weight from 2,475,252 to 2,330,856 pounds, and to reduce individual ice basket ice weight from 1273 to 1199 pounds. The associated Bases section is also revised to reflect the changed requirements.

Date of issuance: August 25, 1998.

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

Amendment Nos.: Unit 1—168; Unit 2—160.

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

Date of initial notice in **Federal Register**: May 6, 1998 (63 FR 25107).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 25, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: December 11, 1997.

Brief description of amendments: The amendments revise Technical Specification Table 3.3–4, Engineered Safety Feature Actuation System Instrumentation Trip Setpoints, to require that suction of the Nuclear Service Water System be swapped from Lake Wylie to the Standby Nuclear Service Water Pond at a higher minimum water level of Lake Wylie. Specifically, the amendments change the swap setpoint from greater than or equal to 554.4 feet to greater than or equal to 557.5 feet, and the allowable value from greater than or equal to 552.9 feet to greater than or equal to 555.4 feet.

Date of issuance: August 25, 1998.

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

Amendment Nos.: Unit 1—169; Unit 2—161.

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

Date of initial notice in **Federal Register**: February 11, 1998 (63 FR 6983).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 25, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: September 15, 1997, as supplemented by letters dated March 5, April 27, June 15, July 22, and August 10, 1998.

Brief description of amendments: The amendments revise Technical Specification Figures 3.4–2 and 3.4–3 (pressure-temperature limits curves), Table 4.4–5 (reactor vessel surveillance capsule withdrawal schedule), and Sections 3/4.4.9.3 and 3.5.3 (requirements concerning overpressure protection). The associated Bases are also revised.

Date of issuance: August 28, 1998.

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

Amendment Nos.: Unit 1—170; Unit 2—162.

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

Date of initial notice in **Federal Register**: October 8, 1997 (62 FR 52580); and July 29, 1998 (63 FR 40553).

The March 5, April 27, July 22, and August 10, 1998, letters provided additional information that did not change the scope of the September 15, 1997, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 28, 1998.

No significant hazards consideration commets received: No.

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

Duke Energy Corporation, Docket No. 50–287, Oconee Nuclear Station, Unit 3, Oconee County, South Carolina

Date of application of amendment: July 20, 1998.

Brief description of amendment: The amendment extends, on a one-time basis, Technical Specification Surveillance 4.18.3 for hydraulic and mechanical snubber testing. The tests are required to be performed at a frequency of 18 months, with a maximum allowed frequency of 22 months, 15 days. The amendment extends this to a maximum of 25 months.

Date of Issuance: August 26, 1998.

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

Amendment No.: 229.

Facility Operating License No. DPR–55: The amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: July 27, 1998 (63 FR 40137).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 26, 1998.

No significant hazards consideration comments received: No.

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

Duke Energy Corporation, Docket No. 50–287, Oconee Nuclear Station, Unit 3, Oconee County, South Carolina

Date of application of amendment: July 16, 1998.

Brief description of amendment: The amendment extends, on a one-time basis, during Operating Cycle 17, certain specified Technical Specification surveillances that are required to be performed at a frequency of 18 months from the maximum allowed frequency of 22 months, 15 days, to a maximum of 24 months.

Date of Issuance: August 28, 1998.

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

Amendment No.: 230.

Facility Operating License No. DPR–55: The amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: July 29, 1998 (63 FR 40555)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 28, 1998.

No significant hazards consideration comments received: No.

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

Duquesne Light Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS–1 and BVPS–2), Shippingport, Pennsylvania

Date of application for amendments: December 19, 1997, as supplemented June 16, July 9, and July 15, 1998.

Brief description of amendments: These amendments revise the requirements for the source range neutron flux channels in Modes 2 (Below P–6), 3, 4, and 5 to incorporate the guidance provided in NUREG–1431, the NRC's improved Standard Technical Specifications with some modifications to address plant-specific design features. This change allows (1) the use of alternate detectors provided the required functions are provided, and (2) plant cooldown with inoperable detectors provided the shutdown margin accounts for the temperature change. This change also modifies the BVPS–2 Technical Specification (TS) Table 3.3–1 Channels To Trip and Minimum Channels Operable requirements to 0 and 1, respectively. This portion of the amendment makes these BVPS–2 requirements consistent with the current BVPS–1 requirements. For both BVPS–1 and BVPS–2, TS Table 4.3–1 is modified to include a notation exempting the alternate source range detectors from surveillance testing until they are required for operability.

Date of issuance: August 26, 1998.

Effective date: Both units, effective immediately, to be implemented within 60 days.

Amendment Nos.: 217 and 94.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 11, 1998 (63 FR 11918).

The June 16, July 9, and July 15, 1998, letters provided clarifying information that did not change the initial no significant hazards consideration determination or expand the amendment request beyond the scope of the March 11, 1998, **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 26, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 3, 1998.

Brief description of amendments: Revise the surveillance requirements of TS Section 4.11.2.5.1, Explosive Gas Mixture, to add a reference the St. Lucie Units 1 and 2 Updated Final Safety Analysis Reports for clarification of an alternative monitoring method to be used in the event that continuous monitoring of explosive gas mixtures in the waste decay tanks becomes inoperable.

Date of Issuance: August 10, 1998.

Effective Date: August 10, 1998, and shall be implemented within 30 days of receipt.

Amendment Nos.: 156 and 94.

Facility Operating License Nos. DPR–67 and NPF–16: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 1, 1998 (63 FR 35990).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 10, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 3, 1998.

Brief description of amendment: This amendment revises the TS in three areas. First, the amendment revises TS 3.4.7, Reactor Coolant System-Chemistry, to eliminate the need for sampling of reactor coolant system chemistry in the defueled condition. Second, the amendment revises TS 5.6.1.a.1, Design Features-Fuel Storage-Criticality, to reflect the total uncertainty associated with the unborated criticality analysis previously approved by NRC. And third, the amendment revises TS 6.5.2.9.d, Technical Review Responsibilities, to be consistent with the quality assurance process previously approved by NRC.

Date of Issuance: August 18, 1998.

Effective Date: As of date of issuance, and shall be implemented within 30 days of receipt.

Amendment No.: 95.

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

Date of initial notice in **Federal Register**: April 8, 1998 (63 FR 17224).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 18, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Community College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34981–5596.

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

Date of application for amendment: April 6, 1998.

Brief description of amendment: The amendment changes the Technical Specifications (TSs) by (1) adding a surveillance requirement to verify pressurizer heater capacity to TS 3.4.4, ''Reactor Coolant System—Pressurizer,'' (2) moving the identification of the location of the containment air temperature detectors from the surveillance requirements portion of TS 3.6.1.5, ''Containment Systems—Air Temperature,'' to the TS Bases for Containment Systems, Section 3/4.4.6.1.5, ''Air Temperature,'' and (3) modifying the action statements and surveillance requirements of TS 3.7.1.5, ''Plant Systems—Main Steam Isolation Valves.'' The TS Bases are updated to include the location of containment air temperature detectors and reflect the changes.

Date of issuance: August 21, 1998.

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

Amendment No.: 219.

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

Date of initial notice in **Federal Register**: May 6, 1998 (63 FR 25113).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 21, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 19, 1998, as supplemented on July 1, 1998. The June 19, 1998, submittal superseded in its entirety Northern States Power (NSP) Company's previous letters dated July 26, 1996, and April 11, 1997. NSP letter dated May 5, 1997, was also considered in the staff's review of the amendment request.

Brief description of amendment: The amendment revises Section 3.6.C, Coolant Chemistry, and 3/4.17.B, Control Room Emergency Filtration System, of the Technical Specifications (TS) to establish TS requirements that are consistent with modified analysis inputs used for the evaluation of the radiological consequences of a postulated main steam line break accident, and of a postulated line break in the reactor water cleanup system.

This amendment request was originally noticed in the **Federal Register** on May 6, 1998 (63 FR 25115).

Date of issuance: August 28, 1998.

Effective date: August 28, 1998. Implementation of the license conditions shall be as specified in Appendix C to DPR–22.

Amendment No.: 101.

Facility Operating License No. DPR–22: Amendment revised the License and the Technical Specifications.

Date of publication of individual notice in **Federal Register**: July 28, 1998 (63 FR 40321).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 28, 1998.

No significant hazards consideration comments received: No.

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

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50–277, Peach Bottom Atomic Power Station, Unit No. 2, York County, Pennsylvania

Date of application for amendment: January 17, 1995, as supplemented by letters dated March 30, 1995; July 2, 1996; February 28 and September 22, 1997; and January 23, July 9 and July 29, 1998.

Brief description of amendment: These amendments revise the technical specifications to support the replacement of the Source Range and Intermediate Range Monitors with the Wide Range Neutron Monitoring System.

Date of issuance: August 24, 1998.

Effective date: As of the date of issuance and is to be implemented upon completion of Modification P00271.

Amendment No.: 222.

Operating License No. DPR–44: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 6, 1995 (60 FR 29885)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 24, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: December 18, 1997, as supplemented July 14, 1998.

Brief description of amendments: The amendments revise the Unit 1 and Unit 2 Facility Operating Licenses by modifying or deleting obsolete conditions.

Date of issuance: August 18, 1998.

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

Amendment Nos.: Unit 1–212; Unit 2–153.

Facility Operating License Nos. DPR–57 and NPF–5: Amendments revised the Facility Operating Licenses.

Date of initial notice in **Federal Register**: January 28, 1998 (63 FR 4324).

The July 14, 1998, letter provided clarifying information that did not change the scope of the December 18, 1997, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 18, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia.

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

Date of application for amendments: October 29, 1996, as supplemented February 19, June 20, and October 21, 1997.

Brief description of amendments: The amendments revise the Technical Specifications (TSs) associated with the oscillation power range monitor portion of the digital Power Range Neutron Monitoring system. The TSs associated with the average power range monitor portion of the system were issued on March 21, 1997.

Date of issuance: August 20, 1998.

Effective date: As of the date of issuance to be implemented on each unit prior to the next refueling outage of that unit.

Amendment Nos.: Unit 1–213; Unit 2–154.

Facility Operating License Nos. DPR–57 and NPF–5: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 2, 1997 (62 FR 130).

The letters dated February 19, June 20, and October 21, 1997, provided clarifying information that did not change the scope of the October 29, 1996, 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 August 20, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia.

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: September 16, 1997, as supplemented by letter dated February 23, 1998.

Brief description of amendments: The amendments would allow sleeving of steam generator tubes with sleeves designed by the vendor, ASEA Brown Boveri/Combustion Engineering (ABB/CE). Additionally, the proposed TS amendment would require that sleeves be removed from service upon detection of service-induced degradation, require post weld heat treatment (PWHT) of sleeve welds, and reduce the allowable primary-to-secondary leakage through any one steam generator to 150 gallons per day (gpd).

Date of issuance: August 26, 1998.

Effective date: August 26, 1998, to be implemented 30 days from the date of issuance.

Amendment Nos.: Unit 2–140; Unit 3–132

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

Date of initial notice in **Federal Register**: January 28, 1998 (63 FR 4323).

The February 23, 1998, supplemental letter provided additional clarifying information and did not change the original no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 26, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 9, 1998, as supplemented by letter dated July 8, 1998.

Brief description of amendment: The amendment revises Technical Specification 4.5.2b.1 and its associated Bases to eliminate the requirement to vent the centrifugal charging pump casings.

Date of issuance: August 17, 1998.

Effective date: August 17, 1998, to be implemented within 30 days from the date of issuance.

Amendment No.: 127.

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

Date of initial notice in **Federal Register**: May 6, 1998 (63 FR 25118)

The July 8, 1998, supplemental letter provided additional clarifying information and did not change the staff's original no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 17, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Missouri-Columbia, Elmer Ellis Library, Columbia, Missouri 65201–5149.

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

Date of application for amendments: September 1, 1995, as supplemented April 8, 1996; April 22, 1996; April 23, 1996; November 18, 1997; February 9, 1998; March 25, 1998; May 5, 1998; June 25, 1998; and June 29, 1998.

Brief description of amendments: The proposed action would revise the Technical Specifications (TS) changing the Emergency Diesel Generator (EDG) outage time from 72 hours to 14 days.

Date of issuance: August 26, 1998.

Effective date: August 26, 1998.

Amendment Nos.: 214 and 195.

Facility Operating License Nos. NPF–4 and NPF–7: Amendments revised the Licenses and the Technical Specifications.

Date of initial notice in **Federal Register**: June 17, 1998 (63 FR 33110), which superseded the notice of September 27, 1995 (60 FR 49949).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 26, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 19, 1998, as supplemented July 14, 1998.

Brief Description of amendments: These amendments revise the Licenses and Technical Specifications (TS) to allow the use of a temporary jumper line for providing service water to component cooling water heat exchangers while maintenance is performed on existing service water supply piping. In addition, editorial changes have been made to TS Table 3.7–2, item 3, and to TS Bases Section 3.14.

Date of issuance: August 26, 1998.

Effective date: August 26, 1998.

Amendment Nos.: 216 and 216.

Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the License and Technical Specifications.

Date of initial notice in **Federal Register**: July 14, 1998 (63 FR 38206). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 26, 1998.

No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 2nd day of September 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 98–24130 Filed 9–8–98; 8:45 am]

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