Consequently, I lack the requisite assurance that Marc W. Zuverink will conduct licensed activities in compliance with the Commission's requirements or that the health and safety of the public will be protected if Marc W. Zuverink were permitted at this time to be involved in NRC-licensed activities. Therefore, the public health, safety and interest require that for a period of ten years from the date of this Order, Marc W. Zuverink be prohibited from any involvement in NRC-licensed activities for either: (1) An NRC licensee, or (2) an Agreement State licensee performing licensed activities in areas of NRC jurisdiction in accordance with 10 CFR 15.020. In addition, for a period of five years commencing after the ten year period of prohibition, Mr. Zuverink must notify the NRC of his employment or involvement in NRC-licensed activities to ensure that the NRC can monitor the status of Mr. Zuverink's compliance with the Commission's requirements and his understanding of his commitment to compliance.

V

Accordingly, pursuant to sections 81, 1761b, 161i, 182, and 186 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR 2.202, 10 CFR Part 30, and 10 CFR 150.20, it is hereby ordered that:

1. Marc W. Zuverink is prohibited for a period of ten years from the date of this Order from engaging in NRC-licensed activities. NRC-licensed activities are those activities that are conducted pursuant to a specific or general license issued by the NRC, including, but not limited to, those activities of Agreement State licensees conducted pursuant to the authority granted by 10 CFR 150.20.

2. For a period of five years, after the above ten year period of prohibition has expired, Marc W. Zuverink shall, within 20 days of his acceptance of each employment offer involving NRC-licensed activities or his becoming involved in NRC-licensed activities, as defined in Paragraph V.1 above, provide notice to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555, of the name, address, and telephone number of the employer or the entity where he is, or will be, involved in the NRC-licensed activities. In the first such notification, Marc W. Zuverink shall include a statement of his commitment to compliance with regulatory requirements and the basis as to why the Commission should have confidence that he will now comply with applicable NRC requirements.

The Director, Office of Enforcement, may, in writing, relax or rescind any of the above conditions upon demonstration by Mr. Zuverink of good cause.

VI

In accordance with 10 CFR 2.202, Marc W. Zuverink must, and any other person adversely affected by this Order may, submit an answer to this Order, and may request a hearing on this Order, within 45 days of the date of this Order. The answer may consent to this Order. Unless the answer consents to this Order, the answer shall, in writing and under oath or affirmation, specifically admit or deny each allegation or charge made in this Order and shall set forth the matters of fact and law on which Mr. Zuverink or other person adversely affected relies and the reasons as to why the Order should not have been issued. Any answer or request for a hearing shall be submitted to the Secretary, U.S. Nuclear Regulatory Commission, Attn: Chief, Docketing and Service Section, Washington DC 20555. Copies also shall be sent to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the Regional Administrator, NRC Region III, 801 Warrenville Road, Lisle, Illinois 60632–4531, if the answer or hearing request is by a person other than Mr. Zuverink. If a person other than Mr. Zuverink requests a hearing, that person shall set forth with particularity the manner in which his or her interest is adversely affected by the Order and shall address the criteria set forth in 10 CFR 2.714(d).

If a hearing is requested by Mr. Zuverink or a person whose interest is adversely affected, the Commission will issue an Order designating the time and place of any hearing. If a hearing is held, the issue to be considered at such hearing shall be whether this Order should be sustained. Since Mr. Zuverink is currently in Federal custody, if a hearing is requested, the Commission will not act on the hearing request until Mr. Zuverink is released from Federal custody. If Mr. Zuverink requests a hearing, the hearing request will not be granted unless Mr. Zuverink: (1) Notifies the Secretary, U.S. Nuclear Regulatory Commission, at the address given above, within 20 days of his release from Federal custody, that he has been released from Federal custody; and (2) provides in the notice his then-current address where he can be contacted and a statement that he continues to desire the hearing. A copy of the notice shall also be sent to the Director, Office of Enforcement, and the Assistant General Counsel for Hearings and Enforcement, at the address given above.

In the absence of any request for hearing, the provisions specified in Section V above shall be effective and final 45 days from the date of this Order without further order or proceedings. In the event that Mr. Zuverink makes the sole request for a hearing and fails to comply with the notification requirements above, the provisions specified in Section V above shall be effective and final 20 days after he is released from Federal custody.

Dated at Rockville, Maryland this 27th day June 1995.

For the Nuclear Regulatory Commission.

Hugh L. Thompson, Jr.,

Deputy Executive Director for Nuclear Materials Safety, Safeguards and Operations Support.

[FR Doc. 95–16371 Filed 7–3–95; 8:45 am]

BILLING CODE 7590–01–M

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

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 June 10, 1995, through June 22, 1995. The last biweekly notice was published on June 21, 1995 (60 FR 32359).

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

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration.

Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

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

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

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

By August 4, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Unit Nos. 1, 2, and 3, Maricopa County, Arizona

Date of amendment requests: May 2, 1995.

Description of amendment requests: The proposed amendment would remove from the technical specifications (TS) plant elevations for the minimum water volume required in the spent fuel pool (SFP) and relocate them to site procedures. This proposed TS amendment also includes two changes to correct administrative errors in the TS.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis about 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 eliminates the plant elevations from TS Figure 3.1–1, ''Minimum Borated Water Volumes'' for the SFP. The change is administrative in nature and does not involve any modifications to plant equipment or affected plant operation. The required volume of water in the SFP is identified on the figure and will remain unchanged by this amendment. This request relocates the plant elevations to site procedures where they will be controlled in accordance with the provisions of 10 CFR 50.59.

The removal of the reference to Table 3.8–2 in the Unit 3 TS 3.8.4.1 and adding the word ''containment'' to the Unit [2] TS 4.6.3.1 are administrative change[s] and do not involve any modifications to plant equipment or affect plant operation. These administrative changes do not affect the scope or intent of any test within the TS.

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

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

The proposed change eliminates the plant elevations from TS Figure 3.1–1, ''Minimum Borated Water Volumes'' for the SFP. The change is administrative in nature and does not involve any modifications to plant equipment or affect plant operation. The removal of plant elevations from the figure does not cause any change in the method by which any safety-related system performs its function. The required volume of water in the SFP is identified on the figure and will remain unchanged by this amendment.

The removal of the reference to Table 3.8–2 in the Unit 3 TS 3.8.4.1 and adding the word ''containment'' to the Unit 2 TS 4.6.3.1 are administrative changes and do not involve any modifications to plant equipment or affect plant operation. These administrative changes do not affect the scope or intent of any test within the TS.

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

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

The proposed change eliminates the plant elevations from TS Figure 3.1–1, ''Minimum Borated Water Volumes'' for the SFP. The change is administrative in nature and does not involve any modifications to plant equipment or affect plant operation. The required volume of water in the SFP is identified on the figure and will remain unchanged by this amendment.

The removal of the reference to Table 3.8–2 in the Unit 3 TS 3.8.4.1 and adding the word ''containment'' to the Unit 2 TS 4.6.3.1 are administrative changes and do not involve any modifications to plant equipment or affect plant operation. These administrative changes do not affect the scope or intent of any test within the TS.

Therefore, based upon the above, 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 that 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: Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004.

Attorney for licensees: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072–3999.

NRC Project Director: William H. Bateman.

Baltimore Gas and Electric Company, Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: June 2, 1995.

Description of amendments request: The proposed amendments would revise the pressurizer safety valve setpoint tolerance ''as-found'' acceptance criterion to +2%/−1% for the valve with the lower setpoint (RC–200) and plus or minus 2% for the valve with the upper setpoint (RC–201). The ''as-left'' setpoint tolerance will remain plus or minus 1% for both valves.

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

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

The pressurizer safety valves are used to prevent exceeding the Reactor Coolant System (RCS) pressure safety limit. The proposed change to increase the pressurizer safety valve setpoint tolerance for the ''as-found'' acceptance criteria from [plus or minus]1% to +2%/−1% for the valve with the lower pressure setpoint, and [plus or minus] 2% for the valve with the upper pressure setpoint, does not affect any initiating event. The proposed change does not affect the consequences of the previously evaluated design basis accidents as the new safety valve setpoint tolerances are bounded by the assumptions in the safety analysis. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to increase the ''as-found'' setpoint tolerances does not involve any changes in equipment or the function of these safety valves. The proposed change does not represent a change in the configuration or operation of the plant. The test method for the pressurizer safety valves will remain the same. The increase in the setpoint tolerances does not create any new accident initiator. Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

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

The pressure safety limit for the RCS protects the structural integrity of the system from failure due to overpressurization. The pressurizer safety valves are used to prevent the RCS pressure from exceeding the safety limit. The proposed change to the pressurizer safety valve setpoint tolerances will continue to prevent the RCS pressure from exceeding the design safety limit during any design basis event. 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 amendments request involves no significant hazards consideration.

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

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

NRC Project Director: Ledyard B. Marsh.

Baltimore Gas and Electric Company, Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: June 6, 1995.

Description of amendments request: The proposed amendments would revise the Calvert Cliffs Nuclear Plant Units 1 and 2 Technical Specifications, extending certain 18-month frequency surveillances to a refueling interval (nominally 24 months, not to exceed 30 months). Systems and equipment affected are the Reactor Protective System (RPS), Engineered Safety Features Actuation System (ESFAS), Power-Operated Relief Valve (PORV) actuation instruments, Low Temperature Overpressure Protection (LTOP)-related instruments, Remote Shutdown Panel instruments, Post-Accident Monitoring (PAM) instruments, Containment Sump Level instruments, and Radiation Monitoring instruments.

This amendment request would extend the nominal surveillance interval requirement from 18 months to a refueling interval (nominally 24 months, not to exceed 30 months) for instrument channel calibrations, RPS and ESFAS total bypass function operability verification, RPS and ESFAS time response tests, ESFAS Manual Trip Button channel functional tests, and ESFAS Automatic Actuation Logic Channel Functional Tests. Calvert Cliffs has been operating on a 24-month fuel cycle since July 1987 (Unit 2) and July 1988 (Unit 1), performing some Technical Specification surveillances, such as the ones described here, during mid-cycle outages. The request is the last of a series of proposed license amendments that would eliminate the need for planned mid-cycle outages to perform required surveillances.

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

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

The proposed change would extend surveillance intervals for Reactor Protective System (RPS), Engineered Safety Features Actuation System (ESFAS), Power-Operated Relief Valve (PORV), Low Temperature Overpressure Protection (LTOP), Remote Shutdown, Post-Accident Monitoring (PAM), Radiation Monitoring, and Containment Sump Level Instruments.

The purpose of the RPS is to effect a rapid reactor shutdown if any one or a combination of conditions deviates from a pre-selected operating range. The system functions to protect the core and the Reactor Coolant System pressure boundary. The purpose of the ESFAS is to actuate equipment which protects the public and plant personnel from the accidental release of radioactive fission products if an accident occurs, including a loss-of-coolant incident, main steam line break, or loss of feedwater incident. The safety features function to localize, control mitigate, and terminate such incidents in order to minimize radiation exposure to the general public. The Post-Accident Monitoring instruments provide the Control Room operators with primary information necessary to take manual actions, as necessary, in response to design basis events, and to verify proper system response to plant conditions and operator actions. The purpose of the Remote Shutdown System is to provide plant parameter indications to operators on a Remote Shutdown Panel to be used while placing and maintaining the plant in a safe shutdown condition in the event the Control Room is uninhabitable. The indications are used to verify proper system response to plant conditions and operator actions. The LTOP System protects against Reactor Coolant System overpressurization at low temperatures by a combination of administrative controls and hardware. The hardware includes two Power-Operated Relief Valves with variable pressurizer pressure setpoints when operating in the LTOP operating parameter region. The Containment Sump High Level Alarm System provides an alarm in the Control Room for a containment sump to provide one of the available indications of excessive RCS leakage during normal plant operation. The Containment Area High Range Radiation Monitoring System provides an indication of high radiation levels in containment. The Containment Purge System actuates equipment to prevent the release of radioactive material to the environment in the event of a reactor coolant leak, a shielding failure, or a fuel pin failure when the reactor vessel head is removed.

The instruments in each of the systems described above are designed to be used in response to an accident. Failure of any of these systems is not an initiator for any previously evaluated accident. Therefore, the proposed change would not involve an increase in the probability of an accident previously evaluated.

Many of the instruments addressed in this license amendment request will have or have recently had a new brand of sensor installed. The effect of the increased surveillance interval with the new sensors was analyzed. The new sensors do not effect the physical design description of the plant, any design or functional requirements, or surveillances. The proposed Technical Specification change extending the surveillance interval from 18 months to a refueling interval (nominally 24 months, not to exceed 30 months) does not physically change the plant, change any design or functional requirements, or effect the surveillances themselves. Analysis has shown that no trip setpoints need to be changed, and operator indications will continue to be accurate for control of plant parameters to effect a safe shutdown. The equipment will continue to perform as designed to mitigate the consequences of accidents. Therefore, the proposed change would not involve a significant increase in the consequences of an accident. [* * *]

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

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

The proposed change to increase the interval RPS, ESFAS, PORV, LTOP, Remote Shutdown, PAM, Radiation Monitoring, and Containment Sump Level instrument surveillances from 18 months to a refueling interval (nominally 24 months, not to exceed 30 months) does not involve a significant change in the design or operation of the plant. No hardware is being added to the plant as part of the proposed change. Some detector upgrades in specific plant systems to enhance the performance of those systems have been or will be performed. However, those upgrades were evaluated and deemed acceptable under 10 CFR 50.59 and are not part of this request. The Reactor Protective System, Engineered Safety Features Actuation System, Power-Operated Relief Valve, Low Temperature Overpressure Protection, Containment Sump Level, one Radiation Monitoring actuation setpoints will not be changed. Analysis has shown that the remote shutdown and PAM indications will continue to be accurate. The proposed change will not introduce any new accident initiators. Therefore, this change does not create the possibility of a new or different type of accident from any previously evaluated.

3. Does operation of the facility in accordance with the proposed amendment

involve a significant reduction in a margin of safety?

The impact of the surveillance interval extension request was evaluated for each Technical Specification-related safety function for each of the RPS, ESFAS, PORV, LTOP, Remote Shutdown, PAM, Radiation Monitoring, and Containment Sump Level instruments addressed by this submittal. In all cases, parameters specified in the related accident analysis were determined to be unaffected by the surveillance interval extension, and no accident analyses limits required changes. The Reactor Protective System, Engineered Safety Features Actuation System, Power-Operated Relief Valve, Low Temperature Overpressure Protection, Containment Sump Level, and Radiation Monitoring actuation setpoints will not be changed. Analysis has shown that the remote shutdown and PAM indications will continue to be accurate. The methods for detection of degraded instrument operation have not been changed, and remote shutdown and PAM operator indications will continue to provide adequate accuracy. The methods for detection of degraded instrument operation have not been changed, and remote shutdown and PAM operator indications will continue to provide adequate accuracy.

The proposed change does not affect the operation of the systems involved. The surveillance interval extension will not affect the design of the systems, and methods for detection of degraded instrument operation will continue to identify operation problems between calibrations. 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 amendments request involves no significant hazards consideration.

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

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

NRC Project Director: Ledyard B. Marsh.

Baltimore Gas and Electric Company, Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: June 9, 1995.

Description of amendments request: The proposed amendments revise the Calvert Cliffs Nuclear Power Plant Radiological Effluent Technical Specifications (RETS) consistent with Generic Letter (GL), ''Implementation of Programmatic Controls For Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or the Process Control Program (Generic Letter 89–01),'' dated January 31, 1989, and the Improved Standard Technical Specifications for Combustion Engineering Plants published in NUREG–1432, as modified by Mr. W. T. Russell's letter of October 25, 1993, ''Content of Standard Technical Specifications,'' to the Improved Technical Specification Owners Group Chairpersons. Changes for relocating the procedural details of the current RETS to the Offsite Dose Control Manual (ODCM) has been prepared in accordance with the proposed changes to the Administrative Controls section of the Technical Specifications.

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

The proposed change has been evaluated against the standards in 10 CFR 50.92 and has been determined to not involve a significant hazards consideration, in that operation of the facility in accordance with the proposed amendments:

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

The proposed changes will provide human factor improvements for the Technical Specifications by relocating existing procedural details of the current Radiological Effluent Technical Specifications to the Offsite Dose Control Manual (ODCM). Procedural details for solid radioactive wastes will be relocated to the Process Control Program. The proposed amendment (1) incorporates programmatic controls in the Administrative Controls section of the Technical Specifications that satisfy the requirements of 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, 10 CFR Part 50, Appendix I, and our current Technical Specifications; (2) relocates the existing procedural details in current specifications involving radioactive effluent monitoring instrumentation, the control of liquid and gaseous effluents, equipment requirements for liquid and gaseous effluents, radiological environmental monitoring, and radiological reporting details from the Technical Specifications to the ODCM; (3) simplifies the associated reporting requirements; (4) simplifies the administrative controls for changes to the ODCM; and (5) updates the definitions of the ODCM consistent with these changes.

Relocating existing requirements and eliminating requirements which duplicate regulatory requirements provide Technical Specifications which are easier to use. Because existing requirements are relocated to established programs where changes to those programs are controlled by regulatory requirements, there is no reduction in commitment and adequate control is still maintained. Likewise, the elimination of requirements which duplicate regulatory requirements enhances the usability of the Technical Specifications without reducing commitments. The additional improvements being proposed neither add nor delete requirements, but merely clarify and improve the readability and understanding of the Technical Specifications. Since the requirements remain the same, these changes only affect the method of presentation, and as such, would not affect possible initiating events for accidents previously evaluated or any system functional requirement.

Furthermore, no safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. The changes are unrelated to the initiation and mitigation of accidents and equipment malfunctions addressed in the Updated Final Safety Analysis Report.

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

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

Transferring the procedural details of radiological effluent monitoring and reporting from the Technical Specifications to the ODCM has no impact on plant operation or safety. No safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. No changes to plant components or structures are introduced which could create new accidents or malfunctions not previously evaluated.

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

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

The margin of safety associated with the affected Technical Specifications is to provide assurance that the releases of radioactive materials during actual or potential releases of liquid or gaseous effluents do not exceed the limits of 10 CFR Part 20. This license amendment request relocates the methodology and parameters used to ensure that the 10 CFR Part 20 limits are maintained, but does not change any of these requirements. Thus, no methodology and parameters for controlling radioactive effluent releases will be changed.

The procedural details of the current Radiological Effluent Technical Specifications will be transferred to the ODCM and replaced with programmatic controls consistent with regulatory requirements, including controls on revisions to the ODCM. Thus, no requirements or controls will be reduced.

The proposed revisions to the reporting requirements for Radiological Effluent Release Report and the revision from the old 10 CFR 20.106 requirements to the new 10 CFR 20.1302 have no impact on plant systems, plant operations or accident precursors. The changes to the effluent reporting requirements and the updated reference to 10 CFR 20.1302 do not change either the means of controlling radioactive releases or the effluent release limits. Therefore, there will be no change in the types and amounts of effluents that will be released, nor will there be an increase in individual or cumulative radiation exposures to any member of the public.

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

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

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

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

NRC Project Director: Ledyard B. Marsh.

Carolina Power & Light Company, Docket No. 50–261, H.B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: June 3, 1995.

Description of amendment request: The requested Technical Specification (TS) change clarifies the definition of operability of the charging pumps by adding a footnote to TS Section 3.2.2.a that states that the connectibility of the emergency power sources is not required for charging pump operability.

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

This change request does not involve a significant hazards consideration for the following reasons.

1. The requested change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The requested change clarifies that the emergency power sources are not required for the operability of the charging pumps. Operation of the charging pumps is not considered in the assumptions for initiation of any analyzed accident and is not credited for accident mitigation in any analyzed accidents in the safety analysis report. Therefore, the availability of emergency power sources to the charging pumps does not affect the probability of occurrence or consequences of an analyzed accident in the safety analysis report.

2. The requested change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The requested change clarifies that the emergency power sources are not required for the operability of the charging pumps. The design requirements of the charging pumps to provide reactor coolant inventory and boron inventory control are not changed. The operability of the emergency power source to the charging pumps is not a precursor to any accident scenario. Failure of the charging pumps is bounded by the plant design which strips the charging pumps from the emergency buses under certain conditions. Since the change does not involve changes in the operation of the plant, or physical or equipment changes or involve controls for accident mitigation equipment, the requested change will not create the possibility of new or different kind of accident from any accident previously evaluated.

3. The requested change clarifies that the emergency power sources are not required for the operability of the charging pumps. Since the charging pumps are stripped from the emergency buses in the event of a loss of power and safety injection, emergency power sources to the charging pumps are not guaranteed to mitigate the consequences of an analyzed accident. As a result, no credit is taken for the charging function in analyzed accidents and the margin of safety as described in the safety analysis report is unchanged. Therefore, the requested change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550.

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

NRC Project Director: David B. Matthews.

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

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

Date of amendment request: February 21, 1995.

Description of amendment request: The proposed amendments would revise Byron and Braidwood technical specifications associated with the reactor coolant system (RCS) resistance temperature detectors (RTDs) used to obtain hot and cold leg temperatures. The amendments are required because of proposed modification that will remove the existing RTDs and their associated piping and valves and replace them with dual element fast response RTDs mounted in the thermowells welded directly in the RCS loop piping.

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 modification replaces the existing bypass piping system with thermowell-mounted RTDs. Because the hot leg RTDs are mounted directly in the scoops, temperature measurement inaccuracies caused by imbalances in the flow scoop sample flow are eliminated. The method of measuring coolant temperature with thermowell-mounted fast response RTDs has been analyzed to be at least as effective as the RTD bypass system. With the thermowells welded into the existing RCS hot and cold leg nozzles and the elimination of the bypass piping, the number of pressure boundary welds has been significantly reduced, resulting in a reduced probability of a small break LOCA [Loss of Coolant Accident].

The RTD response time is incorporated in the safety analyses. In particular, RTD response time is modeled in the OT[DELTA]T [Over Temperature Delta Temperature] and OP[DELTA]T [Over Pressure Delta Temperature] trip functions. The overall response time modeled in the safety analyses for the existing RTD bypass piping system is 8 seconds. The overall response time is the elapsed time from the time the temperature change in the RCS exceeds the trip setpoint until the rods are free to fall. More specifically, 6 seconds is modeled as a first order lag term and 2 seconds as pure delay on the reactor trip signal. The 6 second lag term includes such factors as: RTD bypass piping fluid transport delay, RTD bypass piping thermal lag, RTD response time, and RTD electronic filtering. The 2 second delay on reactor trip addresses such factors as electronics delay, trip breakers and gripper release.

Signal conditioning (filtering) of the individual loop [DELTA]T and T_{avg} signals is represented by [time constants utilized in the lag compensator for DELTA T] and [time constant utilized in the measured T_{avg} lag compensator], respectively, in the OT[DELTA]T and OP[DELTA]T equations in Technical Specification Table 2.2–1. With the current bypass manifold system, the filter is not required since the existing RTDs do not respond rapidly to local temperature variances within the reactor coolant loop. The bypass piping and manifold provide adequate mixing of the coolant, eliminating any local temperature variances. Therefore, the values of [time constants utilized in the lag compensator for DELTA T] and [time constant utilized in the measured T_{avg} lag compensator] are currently specified as 0 seconds, effectively turning off the electronic filter. The new fast response RTDs may respond to temperature spikes which are not representative of actual RCS bulk fluid temperature. Signal conditioning may be required to eliminate these temperature spikes. Although, the current Technical Specifications do not provide for any signal conditioning, the 8 second total response time used in safety analyses has sufficient margin to account for a typical 2 second time constant for signal conditioning. Industry experience has shown that a 2 second filter is adequate in eliminating the spikes.

The proposed fast response RTD/ thermowell system also has an overall response time of 8 seconds. However, the time distribution for the parameters differ between the existing and proposed designs. The existing design includes a transport time for RCS fluid to reach the RTD, located in the manifold. The RTDs are directly immersed into the coolant, providing a fast response. The new design no longer has the transport delay. However, because the RTDs are mounted in thermowells, the response time of the RTD/thermowell combination will be increased over the existing system.

The effects of a redistribution of the time responses between the total lag term (pipe transport delay, RTD response and electronic filter delay) and electronics delay term have been evaluated. Westinghouse completed a Safety Evaluation SECL–95–015, ''OT[DELTA]T and OP[DELTA]T Reactor Trip Response Time Safety Evaluation'' to support the revision to the time requirements. The evaluation concludes that, as long as the total response time remains [less than or equal to] 8 seconds, the safety analyses acceptance criteria continue to be met. The OT[DELTA]T and OP[DELTA]T trip functions are unaffected by the change.

The following Updated Final Safety Analysis Report (UFSAR) Chapter 15 events trip on OT[DELTA]T: Loss of Electric Load/ Turbine Trip, Uncontrolled RCCA Bank Withdrawal at Power, CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant, and Inadvertent Opening of a Pressurizer Safety or Relief Valve. In addition, the following events trip on OP[DELTA]T: Steamline Break at Hot Full Power for Core Response, and Steamline Break Superheat Analysis. These events have been reviewed for a change in the distribution of time responses for OT[DELTA]T and OP[DELTA]T. The review concludes that the time response redistribution did not result in a minimum DNBR lower than the safety analyses limit, did not result in a fuel centerline melt, nor did the superheated steam releases change from those currently existing. Therefore, the radiological consequences for these events do not increase as a result of the less restrictive time response breakdown. Thus, the proposed amendment does not result in an increase in the probability or consequences of a previously evaluated accident.

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

The OT[DELTA]T and OP[DELTA]T trip functions are unaffected by the change. Electronic filtering of the RTD signal has been included, changing the dynamic compensation term of OT[DELTA]T and OP[DELTA]T setpoint equations. No other changes to the setpoint equation result from the proposed modification.

The added 7300 hardware is compatible with the existing 7300 electronic hardware now used. All changes to the 7300 protection cabinets have been qualified. The proposed system is functionally equivalent to the existing one. The proposed modification has been reviewed for conformance with the Institute of Electrical and Electronics Engineers (IEEE) 279–1971 criteria, associated General Design Criteria, Regulatory Guides, and other applicable industry standards. The single failure criterion is satisfied by the proposed modification, since the independence of redundant protection sets is maintained. The new RTD/thermowell system meets the equipment seismic and environmental qualification requirements of IEEE standards 344–1975 and 323–1974, respectively. The proposed changes do not affect the protection system capabilities to initiate a reactor trip. The 2 of 4 voting coincidence logic of the protection sets is maintained. Therefore, the proposed modification meets all appropriate IEEE criteria, industry standards and other guidelines.

In addition, the RTD outputs are used for rod control, turbine runback, pressurizer level and other control systems. These control systems receive the signal after it has been processed at the 7300 cabinets and are therefore unaffected by the proposed modification.

The design and installation of the thermowells is in accordance with the American Society of Mechanical Engineers (ASME) Code requirements. However, should a thermowell fail at the RCS pressure boundary, the resulting accident is enveloped by current design basis accident analyses. Thus, implementation of the proposed amendment does not create the possibility of a new or different kind of accident from any of those previously evaluated.

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

The 7300 protection cabinets calculate individual loop [DELTA]T and T_{avg}, based on the output of the RTDs. These values are used in the OT[DELTA]T and OP[DELTA]T reactor protection trip signals. Electronic filtering of the RTD signal will be included, changing the dynamic compensation term of OT[DELTA]T and OP[DELTA]T setpoint equations. No other changes to the setpoint equation result from the proposed modification. Although the total response time used as input into the safety analyses is unaffected by the proposed modification, the distribution of response times between the total lag (pipe transport delay, RTD response and electronic filter delay) and the electronic delay has changed. The UFSAR events which rely on OT[DELTA]T and OP[DELTA]T trips have been evaluated. The evaluation concludes that the safety analyses acceptance criteria continue to be met, since the total response time is consistent with the safety analyses. The OT[DELTA]T and OP[DELTA]T trips function in the same manner to terminate DNB-related transients. The reliability of the reactor protection system is unaffected by this change. Thus, the proposed modification does not involve a significant reduction in margin of safety.

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

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

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

NRC Project Director: Robert A. Capra.

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

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

Date of amendment request: May 17, 1995.

Description of amendment request: The proposed amendment would modify the technical specifications to allow steam generator tubes to be repaired using the tungsten inert gas (TIG) welded sleeve process developed by ABB Combustion Engineering (ABB/ CE), remove the ability to repair steam generator tubes using the Babcock & Wilcox Nuclear Technologies (BWNT) kinetically welded sleeve process, and increase the requirement to inspect the number of sleeved tubes from 3 percent of the total number of sleeved tubes in all four steam generators (SGs) or all sleeved tubes in one steam generator to 20 percent of each sleeve design installed. The proposed amendments would also delete the requirement to conduct additional corrosion testing to establish the design life for the BWNT kinetically welded sleeve in the presence of a crevice.

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 amendment allows the ABB/CE TIG welded tubesheet sleeves and tube support plate sleeves to be used as an alternate tube repair method for Byron and Braidwood Units 1 and 2 Steam Generators (SGs). The sleeve configuration was designed and analyzed in accordance with the criteria of Regulatory Guide (RG) 1.121 and Section III of the ASME Code. Fatigue and stress analyses of the sleeved tube assemblies produce acceptable results for both types of sleeves as documented in ABB/CE Licensing Report CEN–621–P, Revision 00, ''Commonwealth Edison Byron and Braidwood Unit 1 & 2 Steam Generator Tube Repair Using Leak Tight Sleeves, FINAL REPORT,'' April 1995. Mechanical testing has shown that the structural strength of the sleeves under normal, faulted, and upset conditions is within the acceptable limits specified in RG 1.121. Leakage rate testing for the tube sleeves has demonstrated that primary to secondary leakage is not expected during any plant condition. The consequences of leakage through the sleeved region of the tube is fully bounded by the existing steam generator tube rupture (SGTR) analysis included in the Byron and Braidwood Updated Final Safety Analysis Report (UFSAR).

The current Technical Specification 3.4.6.2.c primary to secondary leakage limit of 150 gallons per day (gpd) through any one SG ensures that SG tube integrity is maintained in the event of main steam line break (MSLB) or loss of coolant accident (LOCA). The RG 1.121 criteria for establishing operational leakage rate limits require a plant shutdown based upon a leak-before-break consideration to detect a free span crack before a potential tube rupture. The 150 gpd limit will continue to allow for early leakage detection and require a plant shutdown in the event of the occurrence of an unexpected crack resulting in leakage that exceeds the TS limit.

The sleeves are designed to allow inservice inspection of the pressure retaining portions of the sleeve and parent tube. Inservice inspection is performed on all sleeves following installation to ensure that each sleeve has been properly installed and is structurally sound. Periodic inspections are performed in subsequent refuel outages to monitor sleeve degradation on a sample basis. The eddy current technique used for inspection will be capable of detecting both axial and circumferential flaws. A 20% sample of the sleeves are inspected each refuel outage. In the event that an imperfection exceeding the repair limit is detected an additional 20% sample will be inspected. The inspection scope is expanded to 100% of the sleeves should a repairable defect be found in the second sample. Tubes that contain defects in a sleeve, which exceed the repair limit, will be removed from service. This ensures that sleeve and tube structural integrity is maintained.

The proposed TS change to support the installation of TIG welded sleeves does not adversely impact any previously evaluated design basis accident. The effect of sleeve installation on the performance of the SG was analyzed for heat transfer, flow restriction, and steam generation capacity. The sleeves reduce the risk of primary to secondary leakage in the SG. The installation of ABB/CE sleeve results in a hydraulic flow restriction that is dependent on the number and types of sleeves installed. The reduction in primary system flow rate is a small percentage of the flow rate reduction seen from plugging one tube and is a preferable alternative when considering core margins based on minimum reactor coolant system flow rates. The sleeving installation will result in a resistance to primary coolant flow through the tube for other evaluated accidents. The results of the analyses and testing, as well as industry operating experience, demonstrate that the sleeve assembly is an acceptable means of maintaining tube integrity. In summary, installation of sleeves does not substantially affect the primary system flow rate or the heat transfer capability of the steam generators.

The sleeve sample size has been increased from 3% of the sleeved tubes in all four steam generators to include an eddy current inspection of a minimum of 20% of each sleeve design installed. Increasing the sample size of the sleeves to be inspected will increase the monitoring of tubes using sleeves for any further degradation while they remain in service. If the sample identifies a sleeve with an imperfection of greater than the repair limit, an additional 20% of the sleeves shall be inspected. The sleeves that have identified imperfections of greater than the repair limit shall be removed from service. Increasing the monitoring of the sleeves will assist in the early detection of a tube or sleeve imperfection and limit the probability of occurrence of an accident previously evaluated in the UFSAR.

Installation of the sleeves can be used to repair degraded tubes by returning the condition of the tubes to their original design basis condition for tube integrity and leak tightness during all plant conditions. The tube bundle overall structural and leakage integrity will be increased with the installation of the sleeves reducing the risk of primary to secondary leakage in the SG while maintaining acceptable reactor coolant system flow rates. Therefore sleeving will not increase the probability of occurrence of an accident previously evaluated.

Removal of the BWNT kinetically welded sleeve process as an approved SG tube repair methodology and not completing the additional corrosion testing necessary to establish the design life for the BWNT kinetically welded sleeve in the presence of a crevice will have no affect on plant operations. There are currently no BWNT kinetically welded sleeves installed in the Byron or Braidwood SGs. Had there been, plant operations would have still been bounded by the existing SGTR analysis in the Byron and Braidwood UFSAR.

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

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

The implementation of the proposed sleeving process will not introduce significant or adverse changes to the plant design basis. Stress and fatigue analyses of the repair has shown the ASME Code and RG 1.121 allowable values are met. Implementation of TIG welded sleeving maintains overall tube bundle structural and leakage integrity at a level consistent with that of the originally supplied tubing. Leak and mechanical testing of the sleeves support the conclusions that the sleeve retains both structural and leakage integrity during all conditions. Repair of a tube with a sleeve does not provide a mechanism that result in an accident outside of the area affected by the sleeve.

Any hypothetical accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing SGTR analysis. The SGTR analysis accounts for the installation of sleeves and the impact on current plugging level analyses. The sleeve design does not affect any other component or location of the tube outside of the immediate area repaired.

The current Technical Specification 3.4.6.2.c primary to secondary leakage limit of 150 gpd through any one SG ensures that SG tube integrity is maintained in the event of an MSLB or LOCA. The limit will provide for leakage detection and a plant shutdown in the event of the occurrence of an unexpected single crack resulting in excessive tube leakage. The leakage limit also provides for early detection and a plant shutdown prior to a postulated crack reaching critical crack lengths for MSLB conditions.

Inservice inspections are performed following sleeve installation to ensure proper weld fusion has occurred to maintain structural integrity. The post installation inspection also serves as baseline data to be used for comparison during future inspections. Periodic eddy current inspections monitor the pressure retaining portions of the sleeve and parent tube for degradation. Eddy current techniques will be employed that are sensitive to axial and circumferential degradation.

Increasing the sample size of tubes repaired using either sleeving process during each scheduled inservice inspection will increase the monitoring of these tubes for any further degradation. The improved monitoring and evaluation of the tube and the sleeves assures tube structural integrity is maintained or the tube is removed for service.

Corrosion testing of typical sleeve-tube configurations was performed to evaluate local stresses, sleeve life, and resistance to primary and secondary side corrosion. The tests were performed on stress relieved and as-welded (non-stress relieved) sleeve-tube joints. Using the corrosion test data in conjunction with finite element analyses of the local stress, the stress relieved joint life was determined to be in excess of 40 years. The ABB/CE TIG welded sleeve operating experience in the industry has shown no sleeve failures due to service induced degradation in sleeves that were installed with acceptable inspection results. This experience includes the stress relieved and as-welded sleeve configurations. ComEd will stress relieve all sleeves at Byron and Braidwood as specified in the Technical Report.

Removal of the BWNT kinetically welded sleeve process as an approved SG tube repair methodology and not completing the additional corrosion testing necessary to establish the design life for the BWNT kinetically welded sleeve in the presence of a crevice will not create the possibility of a new or different type of accident from any accident previously evaluated. Repair of an SG tube with a BWNT kinetically welded sleeve would not have provided a mechanism that resulted in an accident outside of the area affected by the sleeve. Any hypothetical accident as a result of potential tube or sleeve degradation in the repaired portion of the tube would have been bounded by the existing SGTR analysis. The SGTR analysis accounts for the installation of sleeves and the impact on current plugging level analyses. The sleeve design does not affect any other component or location of the tube outside of the immediate area repaired. Furthermore, there are currently no BWNT kinetically welded sleeves installed in the Byron or Braidwood SGs.

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

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

The TIG welded sleeving repair of degraded steam generator tubes has been shown by analysis to restore the integrity of the tube bundle to its original design basis condition. The safety factors used in the design of the sleeves for the repair of degraded tubes are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in steam generator design. The design of the ABB/CE SG sleeves has been verified by testing to preclude leakage during normal and postulated accident conditions.

The portions of the installed sleeve assembly which represents the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation, thus satisfying the requirement of RG 1.83. The portion of the SG tube bridged by the sleeve joints is effectively removed from the pressure boundary, and the sleeve then forms the new pressure boundary. The sleeve enhances the safety of the plant by reestablishing the protective boundaries of the steam generator. Keeping the tube in service with the use of a sleeve instead of plugging the tube and removing it from service increases the heat transfer efficiency of the steam generator. During each scheduled inservice inspection, each sleeve inspected and found to have unacceptable degradation shall be removed from service. The effect on the design transients and the accident analyses have been reviewed based on the installation of sleeves equal to the tube plugging level coincident with the minimum reactor coolant flow rate. Evaluation of the installation of sleeves was based on the determination that LOCA evaluations for the licensed minimum reactor coolant flow bound the combined effect of tube plugging and sleeving up to an equivalent of the actual plugging limit. Sleeving results in a fractional amount of the plugging limitation of one tube and is a preferable alternative when considering core margins based on minimum reactor coolant system flow rates. The sleeving installation will result in a resistance to primary coolant flow through the tube. The primary coolant flow through the ruptured tube is reduced by the influence of the installed sleeve, thereby reducing the consequences to the public due to a SGTR event.

A SG sleeve removes an indication of a possible leak source from the reactor coolant system (RCS) pressure boundary, eliminating the potential of a primary-to-secondary leak. The structural integrity of the tube is maintained by the sleeve and sleeve-to-tube joint.

Installation of either tube sheet or tube support plate sleeves will increase the protective boundaries of the steam generators and will not reduce the margin of safety.

Removal of the BWNT kinetically welded sleeve process as an approved SG tube repair methodology and not completing the additional corrosion testing necessary to establish the design life for the BWNT kinetically welded sleeve in the presence of a crevice will not result in a reduction in the margin of safety. There are currently no BWNT kinetically welded sleeves installed in the Byron or Braidwood SGs. SG tube integrity will be maintained by applying an alternate NRC approved repair methodology or removing the SG tube from service by plugging.

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

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

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

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: April 11, 1995.

Description of amendment request: The proposed amendments would allow a one-time extension of specific LaSalle, Units 1 and 2, 18 month Technical Specification Surveillance Requirements to allow surveillance testing to coincide with the LaSalle, Unit 1, seventh refueling outage (L1R07). The shutdown for L1R07 has been rescheduled from September 1995 until early 1996. The proposed extensions apply to: Calibrations and functional testing of isolation actuation instrumentation, emergency core cooling system actuation instrumentation, and recirculation pump trip actuation instrumentation; leakage testing of reactor coolant system isolation valves; inspection of fire rated seals; functional testing of mechanical snubbers; inspections of emergency diesel generators; and testing of batteries, battery chargers, and other electrical components.

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 because:

The proposed change is temporary and allows a one-time extension of specific surveillance requirements for Unit 1 Cycle 7 to allow surveillance testing to coincide with the seventh refueling outage. The proposed surveillance interval extension is short and will not cause a significant reduction in system reliability nor affect the ability of the systems to perform their design function. Current monitoring of plant conditions and continuation of the surveillance testing required during normal plant operation will continue to be performed to ensure conformance with Technical Specification operability requirements. 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 kind of accident from any accident previously evaluated because:

Extending the surveillance interval for the performance of specific testing will not create the possibility of any new or different kind of accidents. No changes are required to any system configurations, plant equipment, or analyses. Therefore, this change will 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 because:

Surveillance interval extensions will not impact any plant safety analyses since the assumptions used will remain unchanged. The safety limits assumed in the accident analyses and the design function of the equipment required to mitigate the consequences of any postulated accidents will not be changed since only the surveillance test interval is being extended. Historical performance generally indicates a high degree of reliability, and surveillance testing performed during normal plant operation will continue to be performed to verify continued Operability of affected systems, structures and components. Therefore, the plant will be maintained within the analyzed limits, and the proposed extension will not significantly reduce the margin of safety.

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

Local Public Document Room location: Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348.

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: May 19, 1995.

Description of amendment request: The proposed amendments would revise the technical specification requirement to verify each fire protection valve is in the correct position at least once per 31 days. The proposed change will retain a monthly visual inspection of the fire protection valves that are accessible during plant operation. However, the interval for visual surveillance of those valves considered not accessible during plant operation will be changed to at least once per 18 months.

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 because: The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the UFSAR [Updated Final Safety Analysis Report]. The proposed change only changes the testing frequency for valves that are inaccessible during power operation. A check of the LaSalle LER database for the entire operating lifetime of LaSalle Units 1 and 2 was performed, and there has not been any instances in which any Technical Specification related Fire Protection valves have been found out of position. Therefore, the change to the frequency of testing will have no affect on the capability of fire suppression water systems, since all Technical Specification fire protection valves, both accessible and inaccessible at power operation, have a plant history of 100% correct valve lineup during monthly surveillances. Additionally, all fire protection valves that are in the fire suppression water flow path are either locked or seal wired in the required position at all times. The change does not impact the probability of any fire or other accident occurrence. Therefore, the proposed change does not cause an 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 because:

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the UFSAR. The proposed change only changes the testing frequency for valves that are inaccessible during power operation. The change to the frequency of testing will have no effect on the capability of fire suppression water systems, since the valves, both accessible and inaccessible at power operation, have a plant lifetime history of 100% correct valve lineup during monthly surveillances. Additionally, these valves are locked or sealed in the required position at all times. The change does not alter the performance of the fire suppression water system, and therefore introduces no new failure modes. With no alteration or degradation to equipment or system operation, the change introduces no new accident or malfunction.

(3) Involve a significant reduction in the margin of safety because:

The proposed change does not reduce the margin as defined in the bases for any Technical Specification. The proposed change only changes the testing frequency for all Technical Specification fire protection valves that are inaccessible during power operation. The plant history of 100% correct valve lineup for the Technical Specification fire protection valves, combined with the fact that these valves are always locked or sealed in the required position ensures that the bases' minimum OPERABILITY requirements of the fire suppression systems are met.

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

Local Public Document Room location: Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348.

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

NRC Project Director: Robert A. Capra.

Commonwealth Edison Company, Docket Nos. 50–295 and 50–304, Zion Nuclear Power Station, Units 1 and 2, Lake County, Illinois

Date of amendment request: May 31, 1995.

Description of amendment request: The proposed amendments would revise the Technical Specifications and incorporate new acceptance criteria for steam generator tubes with degradation in the tubesheet roll expansion region.

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

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

Application of the F* criteria to degraded steam generator tubes will not affect any of the initiators or precursors of any accident previously evaluated. Application of the proposed change will not increase the likelihood that a transient initiating event will occur because transients are initiated by equipment malfunction and/or catastrophic system failure. The proposed change will allow a new criteria to be applied to disposition steam generator tubes that are degraded in the tubesheet roll transition region. The F* criteria specify a minimum length of tubing which must be free from any indication of degradation. Below the F* length, any type or size of indication, including complete circumferential through wall cracking, will not impact the structural integrity of the tube with respect to pull out forces during normal operation or accident conditions, and does not significantly affect the leakage behavior of the tube. While the Zion UFSAR does not specifically address the Feedwater Line Break (FLB) accident, the FLB event was used as the limiting event in the evaluation of the F* criteria. The FLB pressure differential of 2650 psi maximizes the axial loading on the tube for pull out considerations and is bounding. In addition, the close proximity of the tubesheet to the tube will prevent tube rupture or collapse of the tube in the tubesheet span. Because application of the F* criteria will ensure that degraded tubes will provide the same structural integrity as an original undegraded tube during normal operation and accident and accident conditions, the probability of occurrence of an accident previously evaluated is not significantly increased.

Application of the F* criteria will not significantly increase the consequences of any accident previously evaluated. The F* criteria ensure that sufficient length of undegraded tube exists to maintain structural integrity and preclude significant leakage. Due to the proximity of the tubesheet to the tube, any leakage from degradations below the F* length would be negligible and would be well below the Technical Specification limits established for steam generator leakage. Tube rupture as a result of indications below the F* distance is precluded because the tubesheet prevents outward expansion of the tube in response to internal pressure.

The relationship between the tubesheet region leak rate at the most limiting postulated accident conditions relative to that for normal plant operating conditions has been assessed. For the postulated leak source within the roll expansion, increasing the differential pressure on the tube on the tube wall increases the driving head for the leak; however, it also increases the tube to tubesheet loading.

For a leak source below the F* Distance, the maximum assumed pressure differential results in an insignificant leak rate relative to that which could be associated with normal plant operation. This is a result of the increased tube to tubesheet loading associated with the increased differential pressure. Thus for a circumferential indication within the roll expansion that is left in service in accordance with F* criteria, any leakage under accident conditions would be less than that experienced under normal operating conditions. Therefore, any leakage under accident conditions would be less than the existing Technical Specification leakage limit, which is consistent with accident analysis assumptions. Steam generator tube integrity must be maintained under the postulated loss of coolant accident condition of secondary-to-primary differential pressure. Based on tube collapse strength characteristics, the constraint provided to the tube by the tubesheet gives a margin between the tube collapse strength and the limiting secondary-to-primary differential pressure condition, even in the presence of circumferential or axial indications. The maximum secondary to primary differential pressure during a postulated LOCA is 1005 psi. This value is significantly below the residual preload between the tubes and the tube sheet. Therefore, no significant secondary to primary leakage would be expected to occur.

In addition, the proposed changes will not affect the ability to safely shut down the operating unit and mitigate the consequences of an accident because the proposed changes will not necessitate changes to the emergency procedures governing accident conditions or plant recovery.

Administrative and typographical changes are proposed to correct previous grammatical errors, to eliminate a parenthetical note that could cause confusion when applying the proposed requirements, and to make the terminology used in the Bases section consistent with the definitions provided in Specification 4.3.1. Those proposed changes will not increase the probability of occurrence or consequence of any accident previously evaluated.

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

The proposed changes to the Technical Specifications do not involve the addition of any new or different types of safety related equipment nor do they involve the operation of any equipment required for safe operation of the facility in a manner different from those addressed in the UFSAR. No safety related equipment or function will be altered as a result of the proposed changes. Also, the procedures governing normal plant operation and recovery from an accident are not changed by the application of the F* criteria. The F* criteria will allow the use of an alternate method to plugging or sleeving to repair steam generator tubes with degradation in the tubesheet region. The F* criteria ensure that both the structural integrity and leak tight nature of the steam generator tube will be equivalent to the original tube. Since no new failure modes or mechanisms are introduced by the proposed changes, no new or different type of accident is created.

Administrative and typographical changes are proposed to correct previous grammatical errors, to eliminate a parenthetical note that could cause confusion when applying the proposed requirements, and to make the terminology used in the Bases section consistent with the definitions provided in Specification 4.3.1. Those proposed changes will not create the possibility of a new or different kind of accident from those previously evaluated.

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

Plant safety margins are established through Limiting Conditions for Operation (LCOs), limiting safety system settings, and safety limits specified in Technical Specifications. There will be no changes to the LCOs, limiting safety system settings, or the safety limits as a result of the proposed changes. Application of the F* criteria will allow degraded steam generator tubes to be repaired by an alternative method to plugging or sleeving. Steam generator tube plugging decreases the total primary reactor coolant flow rate and heat transfer capability of the steam generator. While steam generator tube sleeving only slightly reduces the reactor coolant flow rate, large numbers of sleeves can have a measurable effect on flow rate and can complicate steam generator tube inspection activities.

Application of the F* criteria will allow a repair method that will restore the integrity of degraded steam generator tubes and will not adversely affect primary system flow rate or heat transfer capability. Application of the F* criteria will preserve the heat transfer capability of the steam generators and will maintain the design margins assumed in the analyses contained in the UFSAR. The alternate repair method will also be less complicated, faster, and will reduce personnel occupational exposure significantly. Based on the above discussion it is concluded that the proposed changes will not significantly reduce a margin of safety.

Administrative and typographical changes are proposed to correct previous grammatical errors, to eliminate a parenthetical note that could cause confusion when applying the proposed requirements, and to make the terminology used in the Bases section consistent with the definitions provided in Specification 4.3.1. Those proposed changes will not impact any margin of safety.

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

Local Public Document Room location: Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085.

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

NRC Project Director: Robert A. Capra.

Entergy Operations, Inc., Docket Nos. 50–313 and 50–368, Arkansas Nuclear One, Unit Nos. 1 and 2 (ANO–1&2), Pope County, Arkansas

Date of amendment request: April 4, 1995.

Description of amendment request: The proposed amendments revise requirements associated with the ventilation system that services both the Unit 1 and Unit 2 control rooms.

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

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

The control room emergency ventilation and air conditioning systems are not initiators of an accident previously evaluated. Extension of the allowable outage time for one inoperable control room emergency air conditioning system from 7 days to 30 days is acceptable based on the low probability of an event occurring that would require control room isolation and a concurrent or subsequent failure of the remaining operable control room emergency air conditioning system. An evaluation using probabilistic safety assessment techniques has shown the frequency of this event to be at an acceptably low level (4.67E–6/yr). The ANO–1 surveillance requirements for the control room emergency ventilation and air conditioning system has been updated for consistency with the ANO–2 requirements and are consistent with RG 1.52, March 1978, Revision 2. The relaxation in the ANO–2 Mode of Applicability for the control room radiation monitoring instrumentation is acceptable based on the fuel handling accident analysis dose consequences. The analysis assumes that the control room emergency ventilation system is actuated during a fuel handling accident in the containment building. This analysis also shows that the dose consequences to the control room operators are acceptable in the event of a fuel handling analysis in the auxiliary building, assuming that the normal control room ventilation system only is in operation. When the unit is in Mode 5 or Mode 6 (with no handling of irradiated fuel in the containment building), no accident condition has been identified that would require the control room emergency ventilation system to actuate due to high radiation. The remainder of the changes have been made for consistency between the ANO–1 and ANO–2 TS and are considered to be administrative in nature.

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

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

The control room emergency ventilation and air conditioning systems are not accident initiators. The proposed changes introduce no new mode of plant operation and no new possibility for an accident is introduced by modifying the ANO–1 surveillance testing requirements for the control room emergency ventilation and air conditioning systems.

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

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

With the exception of the AOT extension and the relaxation of the ANO–2 Mode of Applicability for the control room radiation monitoring instrumentation, all the ANO–1 and ANO–2 changes are considered administrative or more restrictive and are intended to clarify and make consistent the requirements of the control room emergency habitability equipment. Although the AOT extension does involve an incremental reduction in the margin of safety due to a slight increase in the frequency of an event requiring control room isolation, followed by failure of the operable emergency control room chiller, a probabilistic safety assessment has shown this slight increase in frequency (approximately 3.58E–6/yr) to be acceptably low. The relaxation in the ANO–2 Mode of Applicability for the control room radiation monitoring instrumentation is acceptable based on the fuel handling accident analysis dose consequences. The analysis assumes that the control room emergency ventilation system is actuated during a fuel handling accident in the containment building. This analysis also shows that the dose consequences to the control room operators are acceptable in the event of a fuel handling analysis [sic., accident] in the auxiliary building, assuming that the normal control room ventilation system only is in operation. When the unit is in Mode 5 or Mode 6 (with no handling of irradiated fuel in the containment building), no accident condition has been identified that would require the control room emergency ventilation system to actuate due to high radiation.

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

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

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

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

NRC Project Director: William D. Beckner.

Entergy Operations, Inc., Docket Nos. 50–313 and 50–368, Arkansas Nuclear One, Unit Nos. 1 and 2 (ANO–1&2), Pope County, Arkansas

Date of amendment request: April 4, 1995.

Description of amendment request: The proposed amendments delete requirements to perform inservice inspections of reactor coolant pump flywheels at both Unit 1 and Unit 2.

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

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

Missile generation from a reactor coolant pump (RCP) flywheel could damage the reactor coolant system, the containment, or other equipment or systems important to safety. The fracture mechanics analyses conducted to support the change shows that a preexisting crack sized just below detection level will not grow to the flaw size necessary to create flywheel missiles within the life of the plant. This analysis conservatively assumes minimum material properties, maximum flywheel accident speed, location of the flaw in the highest stress area and a number of startup/shutdown cycles eight times greater than expected. Since an existing flaw in the flywheel will not grow to the allowable flaw size under normal operating conditions or to the critical flaw size under LOCA conditions over the life of the plant, elimination of inservice inspections for such cracks during the plant's life will *not* involve a significant increase in the probability of an accident previously considered.

The proposed changes do not increase the amount of radioactive material available for release or modify any systems used for mitigation of such releases during accident conditions. Therefore, these changes do *not* involve a significant increase in the consequences of any accident previously evaluated.

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

The proposed changes will not change the design, configuration, or method of operation of the plant and therefore, will *not* create the possibility of a new or different kind of accident from any previously evaluated.

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

Significant conservatisms have been used for calculating the allowable flaw size, critical flaw size and crack growth rate in the RCP flywheels. These include minimum material properties, maximum flywheel accident speed, location of the flaw in the highest stress area and a number of startup/shutdown cycles eight times greater than expected. Since an existing flaw in the flywheel will not grow to the allowable flaw size under normal operating conditions or to the critical flaw size under LOCA conditions over the life of the plant, elimination of inservice inspections for such cracks during the plant's life will *not* involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: William D. Beckner.

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

Date of amendment request: April 4, 1995.

Description of amendment request: The proposed amendment revises surveillance requirements associated with the main turbine steam valves.

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

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

Modifying the surveillance frequency of the main turbine-generator (MTG) overspeed protection system introduces no new failure mechanism for the machine, so the consequences, of a postulated MTG overspeed event are no different than those previously evaluated.

As explained in NUREG–1366, ''Improvements to Technical Specifications Surveillance Requirements,'' the present surveillance test frequency requirements were developed for fossil units and carried over to nuclear units due to the similarity in design. However, the particulate concentration, phosphate chemistry and higher steam temperatures present in earlier fossil secondary systems, which were major contributing factors to problems identified by these tests, are not present in the Arkansas Nuclear One-Unit 2 (ANO–2) secondary systems. The operating history of turbine valves at ANO–2 is very good, with no failures identified during the performance of overspeed protection system surveillance testing. Therefore, that change does not involve a significant increase in the probability of any accident previously evaluated.

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

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

Because the proposed changes do not alter the design, configuration, or method of operation of the plant, they do *not* create the possibility of a new or different kind of accident from any previously evaluated.

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

These proposed changes do not alter the acceptance of any surveillance requirements, alter any assumptions used in accident analysis, change any actuation setpoints, nor allow operations in any configuration not previously evaluated. This change in surveillance frequency is based on an operating history of the turbine overspeed protection system which indicates that reducing the test frequency will have no adverse impact on the continued safe operation of the unit.

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

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

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

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

NRC Project Director: William D. Beckner.

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

Date of amendment request: May 31, 1995.

Description of amendment request: The proposed amendment would revise the the Technical Specifications (TS) for the Crystal River Unit 3 to facilitate a 24 month operating cycle by changing the surveillance interval for appropriate TS surveillance requirements that are generally performed during a refueling outage. Additionally, the functional description and the ''Allowable Value'' for three Reactor Protection System and one Emergency Feedwater Initiation and Control System setpoints would be revised. The quantitative limits for determining the operational status of the reactor coolant pumps, the main feedwater pumps, and the main turbine would be relocated from the TS to the Final Safety Analysis Report (FSAR). The surveillance associated with the high radiation setpoint for control room isolation would also be changed to reflect that the setpoint is an ''approximate value'' instead of an ''Allowable value''. The current specified surveillance interval for some equipment and systems which were not re-evaluated or which could not be justified by the evaluation process would not be changed.

Specifically:

1. TS Surveillance Requirements (SR) 3.3.1.6, SR 3.3.5.3, SR 3.3.6.1, SR 3.3.9.2, SR 3.3.10.2, SR 3.3.11.3, SR 3.3.17.2, SR 3.3.18.2, and SR 3.9.2.2 would be revised to extend the surveillance frequency from 18 to 24 months. Also, in TS SR 3.3.17.2 a note would be added indicating the frequency for Function 12 is 18 months.

2. In TS Table 3.3.1–1,

(a) the Function for ''Reactor Coolant Pump Power Monitor (RCPPM)'' would be changed to ''Reactor Coolant Pumps,'' and the ''Allowable Value'' column for this function would be revised to delete the quantitative value and to indicate ''More than one pump tripped'',

(b) the Function for ''Main Turbine Trip (Control Oil Pressure)'' would be changed to ''Main Turbine,'' and the Allowable Value is changed to ''Turbine Tripped'' and

(c) the Function for ''Loss of Both Main Feedwater Pumps (Control Oil Pressure)'' would be changed to ''Main Feedwater Pumps,'' and the Allowable Value is changed to ''Both Pumps Tripped''

3. In TS Table 3.3.11–1, Function 1.a would be changed from ''EFW Initiation—Loss of MFW Pumps (Control Oil Pressure)'' to ''EFW Initiation—Main Feedwater Pumps,'' and the Allowable Value is changed to ''Both Pumps Tripped.''

4. In TS SR 3.3.16.3, the CHANNEL CALIBRATION setpoint would be changed from an allowable value to an approximate setpoint.

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

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. The proposed amendment extends the interval between successive refueling outage based surveillances to once every 24 months for those surveillances evaluated herein and, maintains the existing surveillance interval restriction for those systems and equipment not evaluated for extension. The reliability of systems and components relied upon to prevent or mitigate the consequences of accidents previously evaluated is not degraded beyond that obtained from the currently defined refueling outage interval. Assurance of system and equipment availability is maintained. This change does not involve any change to system or equipment configuration. Therefore, this change does not increase the probability of occurrence or the consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment extends the interval between successive refueling outage based surveillances to once every 24 months for those surveillances evaluated herein and maintains the existing surveillance interval restriction for those systems and equipment not evaluated for extension. This change does not involve any change to system or equipment configuration. Therefore, this change is unrelated to the possibility of creating a new or different kind of accident from any previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. The proposed amendment extends the interval between successive refueling outage based surveillances to once every 24 months for the surveillances evaluated herein, and maintains the existing surveillance interval restriction for those systems and equipment not evaluated for extension. The reliability of systems and components is not degraded beyond that obtained from the currently defined refueling outage interval. Assurance of system and equipment availability is maintained.

Therefore, it is concluded that operation of the facility in accordance with the proposed amendment does not involve a significant reduction in a margin of safety. The proposed extension of the refueling outage interval surveillances to once every 24 months does not degrade the reliability of systems and components beyond that obtained from the currently defined refueling outage interval.

Reliable performance of the systems and equipment effected by this change has been demonstrated.

Implementation of the proposed amendment will maintain the required level of assurance of system and equipment availability. The surveillance interval for systems and equipment that have not been evaluated for extension are excluded from this request. Thus, operation of the facility in accordance with the proposed amendment involves no significant hazards considerations.

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: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 32629.

Attorney for licensee: A.H. Stephens, General Counsel, Florida Power Corporation, MAC–A5D, P. O. Box 14042, St. Petersburg, Florida 33733.

NRC Project Director: David B. Matthews.

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

Date of amendment request: May 31, 1995.

Description of amendment request: The proposed amendment would revise the technical specifications (TS) for the Crystal River Nuclear Plant Unit 3 (CR3) relating to the Once Through Steam Generator's (OTSG's) tube inspection acceptance criteria. Currently, the TS specify repair limit for removing steam generator tubes from service based on a structural evaluation of a simplified model of tubes with uniform through wall (T/W) thinning. A recent tube-pull examination at CR3 identified a number of low signal-to-noise (S/N) tube eddy current indications. The licensee indicated that these S/N indications are a substantially different morphology from the model used to develop the current TS inspection and acceptance limit. As a result of the small signal amplitude associated with these S/N indications, they cannot be accurately sized by conventional bobbin coil phase angle. Therefore, the licensee proposed an alternate methodology for dispositioning the S/N indications. The proposed criteria would address both wear and Inter-Granular-Attack (IGA) degradation mechanisms. Crack-like eddy current indications are *not* included within the proposed scope.

Specifically, the licensee proposed to:

A. Revise TS 5.6.2.10.2, page 5.0–14, ''The results of each sample inspection shall be classified into one of the following three categories:'' to read: ''The results of each bobbin coil sample inspection shall be classified into one of the following three categories:''

B. Revise the Note in TS 5.6.2.10.2, page 5.0–14, ''In all inspections, previously degraded tubes whose degradation has not been spanned by a sleeve must exhibit a significant increase in the applicable imperfection size measurement (> +0.5V bobbin coil amplitude increase for S/N indications or >10% further wall penetration for all other imperfections) to be included in the below percentage calculations.''

C. Revise the sentence in TS 5.6.2.10.4.a.2, page 5.0–16, ''Eddy-current* * *as imperfections'' to read: S/N indications with a bobbin coil amplitude < 0.9V are considered imperfections. Other eddy current testing indications below 20% of the nominal tube wall thickness, if detectable, may also be considered as imperfections.

D. Revise TS 5.6.2.10.4.a.4, page 5.0–16, to read:

''Degraded Tube means a tube containing a S/N indication with a bobbin coil amplitude ≥ 0.9V or other imperfection ≥ 20% of the nominal wall thickness caused by degradation except where all such degradation has been spanned by the installation of a sleeve.''

E. Add TS 5.6.2.10.4.a.7 ''Signal-to-Noise (S/N) indication means an indication whose associated bobbin coil amplitude is < 5 times the background noise, excluding indications located in the tube sheet regions or indications determined to be other than a volumetric morphology.''

F. Renumber 5.6.2.10.4.a.7 to 5.6.2.10.4.a.8, and revise to read: Plugging/Sleeving Limit means the imperfection depth at or beyond which the tube shall be restored to serviceability by the installation of a sleeve or removed from service because it may become unserviceable prior to the next inspection. The Limit for S/N indications is equal to a bobbin coil amplitude of 2.5V, an axial extent of 0.33 inches, or a circumferential extent of 0.6 inches. The Limit is equal to 40% of the nominal tube or sleeve wall thickness for other imperfections. No more than 5000 sleeves may be installed in each OTSG.

G. Renumber 5.6.2.10.4.a.8, and 9 to 5.6.2.10.4.a.9 and 10.

H. Revise TS 5.7.2.c.2, page 5.0–29, to read:

Location, bobbin coil amplitude, and axial and circumferential extent (if determined) for each S/N indication and the location and percent of wall thickness penetration for each other indication of an imperfection, and

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 will not significantly increase the probability or consequences of an accident previously evaluated. The relevant accidents are excessive leakage or steam generator tube rupture (as a consequence of MSLB [Main steam Line Break] or otherwise).

RG [Regulatory Guide] 1.121 establishes a standard method for demonstrating structural integrity under worse-than-DBE [design basis Event] conditions. The existing TS is based on this RG. The S/N disposition strategy continues to rely on this guidance. Current TW sizing techniques would allow defects greater than the current TS limit of 40% to remain in service since these techniques do not accurately measure percent wall penetration for small volume indications. The proposed disposition strategy is based in measurable eddy current parameters of voltage, axial extent, and circumferential extent shown to provide a higher confidence that unacceptable flaws are removed from service. Therefore, the probability of a Steam Generator Tube Rupture (SGTR) is not increased and may well be decreased by implementation of this S/N disposition strategy.

The probability of OTSG tube leakage during normal operation or accident conditions is not adversely affected by the proposed S/N disposition strategy. Operating history indicates essentially no primary to secondary leakage through the OTSG tubes at CR–3. Growth rate studies imply this trend could be expected to continue. Therefore, current leakage limits are retained. Small volume indications which might leak during worse-case FWLB [Feedwater Line Break] conditions are addressed in the RG 1.121 evaluation. The disposition strategy ensure these indications are removed from service as part of the inservice inspection. Once detected, the proposed criteria is at least as effective in determining those indications which should be removed from service as are the existing TS limits.

The S/N disposition strategy is an integral part of an overall effort to better address these and similar phenomena in OTSGs.

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

The key 'new or different' accidents addressed in this and similar proposals is the potential for MSLB-induced multiple SGTR or excessive primary-to-secondary leakage during such events. While these events are addressed in CR–3 Emergency Operating Procedures (EOPs), they are beyond those licensed for the facility.

However, as noted above, the probability of MSLB induced multiple SGTR is reduced by more effective screening and plugging/

sleeving criteria. The probability of detection and identification of tubes which should be removed from service is maintained or improved by the S/N disposition strategy. The likelihood of adverse effects from plugging sound tubes is reduced. The operation of the OTSG or related structures, systems or components is otherwise unaffected.

3. The proposed change will not involve a significant reduction to any margin of safety.

The margins of safety defined in RG 1.121, including the required pressure used in the structural analysis, are retained. The probability of detecting degradation is unchanged since bobbin coil methods will continue to be the primary means of initial detection. The probability of leakage remains acceptably small. The proposed S/N disposition strategy is an enhancement to the inservice inspection of OTSG tubing that will provide a higher level of confidence that tubes exceeding the allowable limits are repaired while sound tubes are left in service. Based upon results of the various growth rate studies, the probability of an accident at the end of cycle is essentially the same as the beginning.

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: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 32629.

Attorney for licensee: A. H. Stephens, General Counsel, Florida Power Corporation, MAC–A5D, P. O. Box 14042, St. Petersburg, Florida 33733.

NRC Project Director: David B. Matthews.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Date of amendment request: June 19, 1995.

Description of amendment request: The licensee proposes to change Turkey Point Units 3 and 4 Technical Specifications (TS) by separation of the 24-hour emergency diesel generator (EDG) run and hot restart EDG test from the loss-of-offsite-power load acceptance test. The licensee revised the original amendment request dated March 30, 1995, by letters dated May 5, 1995, and June 19, 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 was previously presented in the **Federal Register** (60 FR 27339, May 23, 1995). The licensee concluded that the proposed license amendments' revisions do not alter the original conclusion that no significant hazards considerations exist pursuant to 10 CFR 50.92.

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 and its revisions involve no significant hazards consideration.

Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199.

Attorney for licensee: J.R. Newman, Esquire, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Project Director: David B. Matthews.

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 amendment request: January 13, 1995, as supplemented by letters dated April 5 and June 20, 1995.

Description of amendment request: The proposed amendments would change the Facility Operating Licenses and their corresponding Appendices A which contain the Technical Specifications (TS) to permit the implementation of the power uprate program at the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The Hatch units are currently licensed for operation at 2436 megawatts thermal (MWt). The proposed changes would redefine the rated thermal power to 2558 MWt, which represents an increase of 5% over the current licensed level in accordance with the generic boiling water reactor (BWR) power uprate program established by the General Electric Company (GE) and approved by the U.S. Nuclear Regulatory Commission (NRC) staff in a letter from W. T. Russell, NRC, to P. W. Marriott, GE, dated September 30, 1991. Implementation of the proposed power uprate at Plant Hatch will result in an increase of steam flow to approximately 106% of the current value but will require no changes to the basic fuel design. Implementation of this proposed power uprate will require minor modifications, such as resetting the safety relief setpoints, as well as the calibration of plant instrumentation to reflect the uprated power. Plant operating, emergency, and other procedure changes will be made where necessary to support uprated operation.

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

A. Rated Thermal Power is increased to 2558 MWt on page 3 of the Unit 1 Operating License, page 4 of the Unit 2 Operating License, and in Section 1.1 (Definitions) of the Units 1 and 2 Technical Specifications.

Evaluation

The changes in the Operating Licenses and Technical Specifications were evaluated and it was determined that the probability (frequency of occurrence) of design basis accidents occurring is not affected by the increased power level, as the regulatory criteria established for plant equipment (e.g., ASME Code, IEEE standards, NEMA standards, Regulatory Guide criteria) will still be complied with at the uprated power level. Scram setpoints (equipment settings that initiate automatic plant shutdowns) will be established such that there is no significant increase in scram frequency due to uprate. No new challenges to safety-related equipment will result from power uprate.

The changes in consequences of hypothetical accidents which would occur from 102% of the uprated power, compared to those previously evaluated, are in all cases insignificant, because the power uprate accident evaluations will not result in exceeding any NRC-approved acceptance limits. Enclosure 4 of Reference 1, General Electric Report NEDC–32405P, ''Power Uprate Safety Analysis for Edwin I. Hatch Plant Units 1 and 2,'' December 1994, investigated the spectrum of hypothetical accidents and transients, and showed the plant's current regulatory criteria are satisfied at power uprate. For example, in the area of core design, the fuel operating limits will still be met at the uprated power level, and fuel reload analyses will show plant transients meet the criteria accepted by the NRC as specified in NEDO–24011, ''GESTAR II.'' Challenges to fuel or emergency core cooling system (ECCS) performance were evaluated (Section 4.2 of NEDC–32405P) and shown to still meet the criteria of 10 [CFR] 50.46 and Appendix K. Challenges to the containment were evaluated (Section 4.1 of NEDC–32405P) and shown to still meet 10 CFR 50 Appendix A, Criterion 38, Long Term Cooling, and Criterion 50, Containment. Radiological release events were evaluated (Section 9.2 of NEDC–32405P) and shown to meet the criteria of 10 CFR 100 (Unit 1 FSAR Chapter 14 and Unit 2 FSAR Chapter 15).

The results of the analyses discussed above demonstrate that operation at the power uprate level does not significantly increase the probability or consequences of an accident previously evaluated.

B. The surveillance test discharge pressure for the standby liquid control pump at 41.2 gpm is increased from 1190 psig to 1201 psig. This value appears in Surveillance Requirement (SR) 3.1.7.7 and the corresponding Bases Section B 3.1.7 in the Unit 1 and Unit 2 Technical Specifications.

Evaluation

Power uprate operation will result in a 30 psi increase in reactor operating pressure. As will be discussed in these proposed changes, several pressure-dependent setpoints (including safety relief valve [SRV] setpoints) will be increased to preserve current margins. Increasing the pressure 11 psi, at which a 41.2 gpm flow rate is developed, assures continued conformance to anticipated transient without scram (ATWS) criteria at uprated conditions. The surveillance test pressure is based on the maximum pressure for an ATWS event during the time period when the standby liquid control pump is in operation. Section 6.5 of NEDC–32405P discusses the capability of these positive displacement pumps. A small increase in the SRV setpoints will have no effect on the rated injection flow to the reactor. This change, therefore, will not increase the probability or consequences of a previously evaluated accident.

C. The reactor vessel steam dome high pressure allowable value for reactor protection system (RPS) instrumentation is increased 31 psi, consistent with the nominal pressure increase for power uprate. The allowable value appears in Section 3.3.1.1, Table 3.3.1.1–1, Function 3, in the Unit 1 and Unit 2 Technical Specifications.

Evaluation

The reactor vessel steam dome high pressure scram limit is increased because the steam dome operating pressure is increased. Operating pressure for uprated power is increased to assure that satisfactory reactor pressure control is maintained. The operating pressure was chosen on the basis of steam line pressure drop characteristics and the steam flow capability of the turbine. Satisfactory reactor pressure control requires an adequate flow margin between the uprated operating condition and the steam flow capability of the turbine control valves at their maximum stroke. An operating dome pressure of 1035 psig, which is 30 psi higher than the current operating dome pressure, is expected. Therefore, the high pressure scram is increased approximately the same amount to preserve existing margins to reactor trips.

The high pressure scram terminates a pressurization transient not terminated by direct scram or high neutron flux scram. The setting is maintained above the nominal reactor vessel operating pressure and below the specified analytical trip limit used in the safety analyses. The revised high pressure scram setpoint will preserve the hierarchy of pressure setpoints. This means that the high pressure scram setpoint will remain below the opening setpoint of the SRVs. The SRV nominal setpoints are also increased 30 psi, as discussed in Item G below. This hierarchy of setpoints provides assurance that the probability of opening more than one SRV without scram intervention is low.

Since the scram function and the current margins to trip avoidance are maintained with revised setpoints, there is no significant increase in the probability or consequences of an accident previously evaluated.

D. The ATWS reactor vessel steam dome high pressure recirculation pump trip (RPT) allowable value is raised 80 psi. The allowable value appears in Section 3.3.4.2, SR 3.3.4.2.3, in the Unit 1 and Unit 2 Technical Specifications.

Evaluation

The ATWS–RPT high pressure setpoint initiates a trip of the recirculation pumps, thereby adding negative reactivity following events in which a scram does not (but should) occur. Section 5.1.3.2 of NEDC–32405P discusses this function in detail.

The current analytical limit for the ATWS-RPT high pressure trip is 1150 psig. This value was increased 30 psi in the power uprate ATWS safety evaluations to account for the 30 psi increase in vessel operating pressure, SRV setpoints, etc. The current allowable value in the Technical Specifications is 1095 psig. This allowable value was not set by the current analytical limit, but by the range of the installed pressure instruments. As part of the power uprate plant changes, these pressure instruments will be replaced to accommodate higher pressure, and the allowable value, in conjunction with the analytical limit used in the safety analysis, will be increased.

Sections 5.1 and 9.3 of NEDC–32405P show the system can adequately perform its ATWS function with the new setpoint. Therefore, the proposed change does not cause a significant increase in the probability or consequences of an accident previously evaluated.

E. The low-low set (LLS) SRV arming pressure allowable value is increased 31 psi, consistent with the increase in operating pressure and high pressure scram allowable value. The LLS arming pressure allowable value appears in Section 3.3.6.3, Table 3.3.6.3–1, Function 1, in the Unit 1 and Unit 2 Technical Specifications.

Evaluation

The allowable value for the LLS SRV high pressure arming setpoint is increased because the high pressure scram setpoint is increased. No changes to the LLS arming logic associated with the SRV tailpipe pressure switches and the LLS opening and closing pressure setpoints are proposed.

The LLS relief logic mitigates the postulated containment loads of subsequent SRV actuations during small or intermediate loss of coolant accidents (LOCAs) by extending the time between actuations. The LLS logic requires two separate signals to arm itself for operation. Specifically, the LLS logic arms when an SRV opens (i.e., tailpipe pressure switch) and reactor pressure concurrently exceeds the scram setpoint. To preserve the hierarchy of pressure setpoints, the high pressure input to the LLS SRV arming logic has the same setpoint as the high pressure scram, thus minimizing the potential for a spurious SRV opening through the LLS logic without occurrence of a reactor scram.

Increasing the arming setpoint is consistent with increasing the high pressure scram setpoint and will not increase the probability or consequences of an accident previously evaluated.

F. Lower the permissible rod line for single-loop operation (SLO) below 45 percent core flow from the 80 percent rod line to the 76 percent rod line. This Technical Specifications limit appears in Section 3.4.1 (Figure 3.4.1–1) and the corresponding Bases Section B 3.4.1 of the Unit 1 and Unit 2 Technical Specifications.

Evaluation

During development of the generic power uprate program, GE and the NRC agreed to maintain the current exclusion region in the power-to-flow map related to thermal-hydraulic stability. The current limit for SLO is the 80 percent rod line. Power uprate will redefine 100 percent rated power and, therefore, rated rod or flow control lines. The 76 percent rod line at uprated conditions closely corresponds on an absolute, rather than percentage basis, to the existing 80 percent rod line.

Therefore, this proposed Technical Specifications change ensures that power uprate operation will not cause a significant increase in the probability or consequences of accident previously evaluated.

G. The SRV lift setpoints in the Units 1 and 2 Technical Specifications SR 3.4.3.1 will be increased 30 psi.

Evaluation

The SRVs are designed to prevent overpressurization of the reactor pressure vessel during abnormal operational transients. The SRV lift setpoints are increased to accommodate the increase in operating pressure that accompanies power uprate. The increase in SRV setpoints ensures that adequate margins are maintained so that the increase in dome pressure during normal operation does not result in an increase in the number of unnecessary SRV actuations. The setpoint increase also maintains the hierarchy of pressure setpoints described in these proposed changes. Transient evaluations include a +3 percent tolerance to the nominal setpoints. As described in Section 3.2 of NEDC–32405P, peak vessel pressure increases by 3 percent, but remains well below the 1375 psig ASME Code limit.

Although not credited in the transient analysis, GPC installed a pressure transmitter system which can electronically actuate the SRVs on high vessel pressure. The nominal trip setpoints for its actuation correspond with the nominal mechanical lift setpoints in the Technical Specifications. The SRV pressure transmitter system nominal setpoints will also be increased 30 psi.

General Electric generically evaluated the adequacy of BWR SRVs to operate at uprated temperatures and pressures. The reactor operating pressure and temperature increases of less than 40 psi and 5°F, respectively, used in that evaluation bound the uprated Hatch operating conditions.

The impact of power uprate on the Hatch containment dynamic loads due to SRV discharge has also been evaluated. As discussed in Section 4.1.2 of NEDC–32405P, the vent thrust loads with power uprate were calculated to be less than the loads used in the containment analysis. The effects of power uprate on SRV air-clearing, the discharge line, the pool pressure boundary, and submerged structure drag loads are discussed in Section 4.1.2 of NEDC–32405P which concludes that the small increase in the setpoint pressure is well within the margin in the SRV loads defined in the Mark I Containment Long-Term Program. Therefore, power uprate does not impact the Hatch SRV load definitions used in the containment analysis, and no significant increase in the probability or consequences of an accident previously evaluated is caused by this proposed change.

H. The Limiting Condition for Operation (LCO) and SRs for the maximum reactor steam dome pressure will be increased from 1020 psig to 1058 psig. This requirement appears in LCO 3.4.10, SR 3.4.10.1, and the corresponding Bases in the Unit 1 and Unit 2 Technical Specifications.

Evaluation

As discussed in the Technical Specifications Bases and NEDC–32405P, the maximum reactor dome pressure is an initial condition of the vessel overpressure protection analysis, which assumes a fast isolation of all four main steam lines by the main steam isolation valves (MSIVs). The reactor scram signal generated directly by the valve closure is assumed defeated for this analysis. Instead, the scram signal is generated by high neutron flux. The overpressure analysis for power uprate assumed an initial dome pressure of 1058 psig, which represents an increase of 38 psig. This initial pressure was chosen approximately 2 percent above the 1035 psig steam dome operating pressure expected for power uprate operation. The analysis also included the other changes (including SRV setpoints) discussed in these proposed changes. Therefore, there is no significant increase in the probability or consequences of an accident previously evaluated.

I. The HPCI and RCIC surveillance test pressures in Units 1 and 2 Technical Specifications SRs 3.5.1.8 and 3.5.3.3, respectively, are increased 38 psi.

Evaluation

The allowable HPCI and RCIC surveillance test pressure is increased to correspond with the increase in normal reactor operating pressure and LCO/SR on maximum reactor pressure that accompanies power uprate. (As discussed in Item H above, the LCO on reactor steam dome pressure is increased 38 psi.) The change is needed to ensure that pressure and power reductions are not required to perform surveillance testing. The requested changes will allow the quarterly demonstration of the HPCI and RCIC systems' capability to perform at normal reactor operating pressures, which meets the original intent of the Technical Specifications.

The HPCI and RCIC systems have been evaluated and demonstrated to be capable of injecting design flow rate at the higher reactor pressure as discussed in Sections 4.2 and 3.8 of NEDC–32405P and in Reference 2.

Therefore, these changes will ensure that power uprate operation will not cause a significant increase in the probability or consequences of an accident previously evaluated.

J. Bases Changes

Several changes to the Hatch Units 1 and 2 Technical Specifications Bases are proposed for consistency with the power uprate safety analyses. These proposed changes are in addition to the Bases changes corresponding to proposed changes A through I.

i. The main steam line flow differential pressure setpoints (Bases Section B 3.3.6.1.c) and the HPCI/RCIC high flow differential pressure setpoints (Bases Section B 3.3.6.3.a and B 3.3.6.4.a) are changed for both units.

The allowable values (in percent of rated) will not change for power uprate operation. However, the actual differential pressure will change due to the increase in steam flow and pressure.

ii. The HPCI and RCIC upper design pressure in Bases Sections B 3.5.1 and B 3.5.3, respectively, is increased 34 psi for both units

The Bases changes support the design of these high pressure systems to pump rated flow from approximately 150 psig up to a pressure associated with the first group of SRV setpoints. This proposed design pressure conservatively considers the 30 psi higher nominal setpoints and 3 percent setpoint drift. The capability of the HPCI and RCIC systems to deliver design flows at these pressures is discussed in Reference 2, and was reviewed by GE for the Unit 1 and Unit 2 systems.

Note that the upper design pressure for HPCI and RCIC is different from the surveillance test pressure for HPCI and RCIC discussed previously in item I. The maximum surveillance test pressure corresponds to reactor operating pressure, since the surveillance test is performed when the unit is operating. The HPCI and RCIC upper design pressure reflects the capability to inject water to the vessel following a reactor scram and isolation.

iii. The peak post accident containment pressure (P_a) is changed to 49.6 psig (Unit 1) and 45.5 psig (Unit 2). These values appear in Bases Sections B 3.6.1.1, B 3.6.1.2, and B 3.6.1.4 in each unit's Technical Specifications.

Section 4.1.1.3 of NEDC–32405P discusses the peak short-term containment pressure response which was recalculated for power uprate conditions. Containment pressure and temperatures remain below design limits and are essentially unchanged.

iv. The main condenser offgas gross gamma activity rate limit of 240 mci/second will not be changed for power uprate. A statement that the current limit is conservative for power uprate conditions was added to Bases Section 3.7.6 for both units.

The Bases derive the current 240 mci/second limit using a rated core thermal power limit of 2436 MWt. A slightly higher limit could be justified using the uprated power level. However, adequate margin exists with the current limit.

v. The inservice hydrostatic and leak testing pressures shown in Bases Section 3.10.1 are increased 33 psi and 30 psi, respectively. This change affects each unit's Bases.

This change is a direct result of the 30 psi increase in normal operating pressure proposed for power uprate. The leakage test is normally performed at operating pressure and the hydrostatic test at approximately 110 percent of operating pressure.

The above Bases changes Items i–v have been evaluated and will not increase the probability or consequences of an accident previously evaluated.

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

Evaluation

The Operating License changes in power level and the associated Technical Specifications changes discussed previously will not create the possibility of a new or different kind of accident from any accident previously evaluated, as summarized below.

Equipment that could be affected by power uprate was evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode were identified. The full spectrum of accident considerations defined in RG 1.70 was evaluated, and no new or different kind of accident was identified. Uprate uses already-developed technology and applies it within the capabilities of existing plant equipment in accordance with presently existing regulatory criteria to include NRC-approved codes, standards, and methods. GE has designed BWRs of higher power levels than the uprated power of any of the currently operating BWR fleet, and no new power dependent accidents have been identified.

The Technical Specifications changes required to implement power uprate require only minor modifications to the plant's configuration. All changes were evaluated and found to be acceptable.

3. Will the changes involve a significant reduction in the margin of safety?

A. Rated Thermal Power is increased to 2558 MWt on page 3 of the Unit 1 Operating License, page 4 of the Unit 2 Operating License, and in Section 1.1 (Definitions) of the Unit 1 and Unit 2 Technical Specifications.

Evaluation

The events analyzed in the FSAR were re-evaluated to demonstrate that power uprate can be implemented without exceeding any regulatory limit. Because the applicable safety analysis criteria and limits are satisfied for power uprate, the margin of safety associated with the safety limits and other limits identified in the Technical Specifications will be maintained.

As discussed in NEDC–32405P, the safety margins prescribed by the Code of Federal Regulations are maintained by meeting the appropriate regulatory criteria. Similarly, the margins provided by the application of the ASME design criteria are maintained. Section 11.4.2 of NEDC–32405P discusses the effects of power uprate on safety margins for the following:

Fuel thermal limits Design basis accidents and the challenges to fuel, containment, and radiological releases. Transient analyses. Non-LOCA radiological releases. Environmental consequences.

These evaluations conclude that applicable safety analysis criteria and limits are satisfied, and thus, the margin of safety will not be significantly reduced.

B. The surveillance test discharge pressure for the SLC pump at 41.2 gpm is increased from 1190 psig to 1201 psig. This value appears in SR 3.1.7.7 and corresponding Bases Section B 3.1.7 in the Unit 1 and Unit 2 Technical Specifications.

Evaluation

Power uprate operation will result in a 30 psi increase in reactor operating pressure. Several pressure-dependent setpoints (including SRV setpoints) will be increased to preserve current margins. Increasing the pressure 11 psi, at which a 41.2 gpm flow rate is developed, assures continued conformance to ATWS criteria at uprated conditions. The surveillance test pressure is based on the maximum pressure for an ATWS event during the time period when the SLC pump is in operation. Section 6.5 of NEDC–32405P discusses the capability of these positive displacement pumps. A small increase in the SRV setpoints will have no effect on the rated injection flow to the reactor.

For power uprate, the capability of the SLCS to respond with adequate margin to an ATWS event was confirmed. The results are reported in Section 9.3.1 of NEDC–32405P. The limiting ATWS event was an inadvertent MSIV closure. The event was reanalyzed at uprate conditions with the higher SRV setpoints and ATWS-RPT setpoints. Peak vessel pressure was well below the ASME emergency limit of 1500 psig. The effect of power uprate on peak clad temperature and maximum suppression pool temperature was judged to be negligible, because the calculations showed no increase in fuel surface heat flux or integrated SRV flow.

In summary, all ATWS criteria are satisfied and the SLC pumps are capable of injecting the required amounts of sodium pentaborate at uprated conditions. Therefore, there is no significant decrease in the margin of safety.

C. The reactor vessel steam dome high pressure allowable value for RPS instrumentation is increased 31 psi, consistent with the nominal pressure increase for power uprate. The allowable value appears in Section 3.3.1.1, Table 3.3.1.1–1, Function 3, in the Unit 1 and Unit 2 Technical Specifications.

Evaluation

The reactor vessel steam dome high pressure scram limit is increased because the steam dome operating pressure is increased. Operating pressure for uprated power is increased to assure that satisfactory reactor pressure control is maintained. The operating pressure was chosen on the basis of steam line pressure drop characteristics and the steam flow capability of the turbine. Satisfactory reactor pressure control requires an adequate flow margin between the uprated operating condition and the steam flow capability of the turbine control valves at maximum stroke. An operating dome pressure of 1035 psig, which is 30 psi higher than the current operating dome pressure, is expected. Therefore, the high pressure scram is increased approximately the same amount to preserve existing margins to reactor trips.

The increases in the steam dome high pressure scram instrument setpoints for uprated power were evaluated by determining whether the high pressure scram, which is used as a backup to other scram signals, provides adequate overpressure protection. The evaluation demonstrates that the backup protection function, with the revised setpoints, continues to provide adequate overpressure protection at uprated power conditions by meeting the applicable ASME Code criteria. Therefore, there is no significant decrease in the margin of safety.

D. The ATWS reactor vessel steam dome high pressure RPT allowable value is raised 80 psi. The allowable value appears in Section 3.3.4.2, SR 3.3.4.2.3, in the Unit 1 and Unit 2 Technical Specifications.

Evaluation

The ATWS-RPT high pressure setpoint initiates a trip of the recirculation pumps, thereby adding negative reactivity following events in which a scram does not (but should) occur. Section 5.1.3.2 of NEDC–32405P discusses this function in detail.

For power uprate, the capability of the SLCS to respond to a postulated ATWS event with adequate margin was confirmed (Section 9.3.1 of NEDC–32405P). By reducing reactor power until the SLCS can inject the required amounts of sodium pentoborate to achieve full shutdown, the RPT also reduces suppression pool temperature for isolation cases (also shown to be acceptable for power uprate conditions in Section 9.3.1 of NEDC–32405P). Therefore, there is no significant decrease in a margin of safety.

E. The LLS SRV arming pressure allowable value is increased 31 psi, consistent with the increase in operating pressure and high pressure scram allowable value. The LLS arming pressure allowable value appears in Section 3.3.6.3, Table 3.3.6.3–1, Function 1, in the Unit 1 and Unit 2 Technical Specifications.

Evaluation

The allowable value for the LLS SRV high pressure arming setpoint is increased, because the high pressure scram setpoint is increased. No changes to the LLS arming logic associated with the SRV tailpipe pressure switches, and the LLS opening and closing pressure setpoints are proposed.

Since this proposed change only affects one of two arming signals for LLS, the safety analyses are not affected; therefore, there is not a significant change in the margin of safety.

F. Lower the permissible rod line for SLO below 45 percent core flow from the 80 percent rod line to the 76 percent rod line. This Technical Specifications limit appears in Section 3.4.1 (Figure 3.4.1–1) and corresponding Bases Section B 3.4.1 of the Unit 1 and Unit 2 Technical Specifications.

Evaluation

This change to the power versus flow map restricted zone is made to maintain the same operating constraints and stability margin that were established for the current power level. This change avoids any increase in the possibility of occurrence or any increase in the potential effects of power oscillations. Therefore, there is no significant decrease in a margin of safety.

G. The SRV lift setpoints in Surveillance Requirement 3.4.3.1 (both units) will be increased 30 psi.

Evaluation

The SRVs are designed to prevent overpressurization of the reactor pressure vessel during abnormal operational transients. The SRV lift setpoints are increased to accommodate the increase in operating pressure that accompanies power uprate. The increase in SRV setpoints ensures that adequate margins are maintained so that the increase in dome pressure during normal operation does not result in an increase in the number of unnecessary SRV actuations. The setpoint increase also maintains the hierarchy of pressure setpoints described in these proposed changes. Transient evaluations include a + 3 percent tolerance to the nominal setpoints. As described in Section 3.2 of NEDC–32405P, peak vessel pressure increases by 3 percent but remains well below the 1375 psig ASME Code limit. Therefore, there is no significant decrease in the margin of safety.

H. The Limiting Condition for Operation (LCO) and Surveillance Requirements for the maximum reactor steam dome pressure will be increased from 1020 psig to 1058 psig. This requirement appears in LCO 3.4.10, SR 3.4.10.1, and the corresponding Bases in the Unit 1 and Unit 2 Technical Specifications.

Evaluation

As discussed in the Technical Specifications Bases and in Section 3.2 of NEDC–32405P, the maximum reactor dome pressure is an initial condition of the vessel overpressure protection analysis, which assumes a fast isolation of all four main steam lines by the main steam isolation valves. It is also used as a sensitivity study parameter for certain transient and LOCA events.

With this revised limit, peak vessel pressure remains below ASME Code criteria, transient limits are maintained, and LOCA fuel performance satisfies the requirements of 10 CFR 50.46 and 10 CFR 50, Appendix K. Therefore, there is no significant decrease in a margin of safety.

I. The HPCI and RCIC surveillance test pressures in SRs 3.5.1.8 and 3.5.3.3, respectively, (both units) are increased 38 psi.

Evaluation

The allowable HPCI and RCIC surveillance test pressure is increased to correspond with the increase in normal reactor operating pressure and LCO/SR on maximum reactor pressure that accompanies power uprate. (As discussed previously, the LCO on reactor steam dome pressure is increased 38 psi.)

The purpose of the HPCI and RCIC surveillance test is to provide periodic demonstration of the systems' ability to perform consistent with the requirements of the analyses at the higher operating pressure associated with power uprate conditions. An evaluation of the HPCI and RCIC systems confirmed their ability to operate at slightly higher turbine speed and provide design flow at power uprate conditions. System performance will be confirmed during the initial power ascension to uprated conditions (and periodically thereafter per the Technical Specifications). Therefore, there is no significant decrease in the margin of safety.

J. Bases Changes

Several changes to the Hatch Units 1 and 2 Technical Specifications Bases are proposed for consistency with the power uprate safety analyses. These proposed changes are in addition to the Bases changes corresponding to proposed changes A through I.

i. The main steam line flow differential pressure setpoints, as shown in Bases Section B 3.3.6.1.c, and the HPCI/RCIC high flow differential pressure setpoints (Units 1 and 2 Bases Sections B 3.3.6.3.a and B 3.3.6.4.a) are changed.

The allowable values (in percent of rated) will not change for power uprate operation. However, the actual differential pressure will change due to the increase in steam flow and pressure.

ii. The HPCI and RCIC upper design pressure in Units 1 and 2 Bases Sections B 3.5.1 and B 3.5.3, respectively, is increased 34 psi.

The Bases changes support the design of these high pressure systems to pump rated flow from approximately 150 psig up to a pressure associated with the first group of SRV setpoints. This proposed design pressure conservatively considers the 30 psi higher nominal setpoints and 3 percent setpoint drift. The capability of the Unit 1 and Unit 2 HPCI and RCIC systems to deliver design flows at these pressures was reviewed by GE and is discussed in Reference 2.

iii. The peak post accident containment pressure (P_a) is changed to 49.6 psig (Unit 1) and 45.5 psig (Unit 2). These values appear in Units 1 and 2 Bases Sections B 3.6.1.1, B 3.6.1.2, and B 3.6.1.4.

Section 4.1.1.3 of NEDC–32405P discusses the peak short-term containment pressure response which was recalculated for power uprate conditions. Containment pressure and temperatures remain below design limits and are essentially unchanged.

iv. The main condenser offgas gross gamma activity rate limit of 240 mci/second will not be changed for power uprate. A statement that the current limit is conservative for power uprate conditions was added to Units 1 and 2 Bases Section 3.7.6.

The Bases derive the current 240 mci/second limit using a rated core thermal power limit of 2436 MWt. A slightly higher limit could be justified using the uprated power level. However, adequate margin exists with the current limit.

v. The inservice hydrostatic and leak testing pressures shown in Units 1 and 2 Bases Section 3.10.1 are increased 33 psi and 30 psi, respectively.

This change is a direct result of the 30 psi increase in normal operating pressure proposed for power uprate. The leakage test is normally performed at operating pressure and the hydrostatic test at approximately 110 percent of operating pressure.

The above Bases changes i–v were evaluated, and there is no significant decrease 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: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513.

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

NRC Project Director: Herbert N. Berkow.

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50–366, Edwin I. Hatch Nuclear Plant, Unit 2, Appling County, Georgia

Date of amendment request: April 14, 1995.

Description of amendment request: The licensee proposes to revise Plant Hatch Unit 2 Technical Specifications (TS) to eliminate selected response time testing requirements from the TS. Specifically, the response time testing to be eliminated includes sensors and specified loop instrumentation for: (1) the Reactor Protection System, (2) the Isolation System, and (3) the Emergency Core Cooling System (ECCS). The deletion of instrumentation from the ECCS response time testing necessitates moving the remaining portion of the test to the ECCS system TS. In addition, the Note for Surveillance Requirement 3.3.6.1.7, which reads: ''Radiation detectors may be excluded,'' is being removed since response time testing is not required for any radiation detector that provides a primary containment isolation signal as indicated in Table 3.3.6.1–1.

Proposed TS Changes 1, 2, and 3 are supported by an analysis performed by the BWR Owners' Group (BWROG), with the licensee's participation. The analysis was submitted to the NRC for approval as Topical Report NEDO–32291, ''System Analyses for the Elimination of Selected Response Time Testing Requirements,'' Boiling Water Reactor Owners' Group, January 1994. The NRC approved the Topical Report by a Safety Evaluation Report (SER) issued on December 28, 1994, ''Evaluation of Boiling Water Reactor Owners' Group Topical Report NEDO–32291, System Analyses for the Elimination of Selected Response Time Testing Requirements.'' The BWROG analysis demonstrates that other periodic tests required by TS, such as channel calibrations, channel checks, channel functional tests, and logic system functional tests, ensure that instrument response times are within acceptable limits. The applicability of the referenced analysis to Plant Hatch has been verified. Proposed Change 4 removes an unnecessary note, since no functions subject to this surveillance include radiation monitors.

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:

Basis for Proposed Changes 1, 2, and 3

1. The changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The purpose of the proposed changes is to eliminate response time testing requirements for selected instrumentation in the RPS [Reactor Protection System], Isolation System], and ECCS. However, because of the continued application of other existing Technical Specifications requirements, such as channel calibrations, channel checks, channel functional tests, and logic system functional tests, the response time of these systems will be maintained within the acceptance limits assumed in plant safety analyses. This will assure successful mitigation of an initiating event. The proposed Technical Specifications changes do not affect the capability of the associated systems to perform their intended function within their required response time.

The BWR Owners' Group (BWROG) has documented an evaluation in NEDO–32291, ''System Analyses for Elimination of Selected Response Time Testing Requirements,'' which was submitted to the NRC for review and approval as a Topical Report in January 1994 and subsequently approved by an NRC SER in December 1994. This evaluation demonstrates that response time testing is redundant to the other Technical Specifications requirements listed in the preceding paragraph. These other tests are sufficient to identify failure modes or degradation in instrument response time and ensure operation of the associated systems within acceptance limits. There are no known failure modes that can be detected by response time testing that cannot also be detected by the other Technical Specifications tests.

2. The proposed changes will not create the possibility of a new or different kind of accident from any accident previously analyzed. As discussed above, the proposed Technical Specifications changes do not affect the capability of the associated systems to perform their intended function within the acceptance limits assumed in plant safety analyses.

3. The proposed changes do not involve a significant reduction in the margin of safety. The current Technical Specifications response times are based on the maximum allowable values assumed in the plant safety

analyses, which conservatively establish the margin of safety. As described above, the proposed Technical Specifications changes do not affect the capability of the associated systems to perform their intended function within the allowed response time used as the basis for the plant safety analyses. Plant and system responses to an initiating event will remain in compliance with the assumptions of the safety analyses; therefore, the margin of safety is not affected.

Although not explicitly evaluated, the proposed Technical Specifications changes enhance plant safety and operation by:

a. Reducing the time safety systems are unavailable,

b. Reducing safety system actuations,

c. Reducing shutdown risk,

d. Limiting radiation exposure to plant personnel, and

e. Eliminating the diversion of key personnel to conduct unnecessary testing.

Basis for Proposed Change 4

1. The change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The Note for SR 3.3.6.1.7 indicates that response time testing for radiation detectors that provide primary containment isolation signals as indicated in Table 3.3.6.1–1 is not required. However,

Table 3.3.6.1–1 does not reference SR 3.3.5.1.7 for any radiation detector that provides primary containment isolation signals. The proposed change eliminates the potential for confusion during instrumentation surveillance testing. Deletion of the note will not prevent the radiation detectors from performing their intended function and will not affect the results of any accident analysis.

2. The proposed changes will not create the possibility of a new or different kind of accident from any accident previously analyzed. As discussed above, the proposed Technical Specifications change eliminates the potential for confusion during instrumentation surveillance testing. This change does not modify any plant equipment or change any plant procedure that provides instructions for the operation of plant equipment. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously analyzed.

3. The proposed change does not involve a significant reduction in the margin of safety. The Note that is being deleted by the change states that testing is not required for instrument sensors which is not required by the SR. Therefore, the Note is superfluous and could cause confusion during instrumentation surveillance testing. The proposed change eliminates that potential. This change is conservative, since it deletes a statement that was intended to reduce the amount of surveillance testing performed on certain instrumentation. The proposed change does not affect plant equipment, procedures, or radiation release prevention and mitigating functions. 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: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513.

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: March 17, 1995.

Description of amendment request: The amendments would revise Technical Specification (TS) 3.9.4, Containment Building Penetrations, to allow the personnel airlock to be open during core alterations or movement of irradiated fuel within the containment.

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 to the Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change to Specification 3.9.4 would allow the containment personnel airlock (PAL) to be open during fuel movement and core alterations. The PAL is currently closed during fuel movement and core alterations to prevent the escape of radioactive material in the event of a fuel handling accident. The PAL is not an initiator to any accident. Whether the PAL doors are opened or closed during fuel movement or core alterations has no effect on the probability of any accident previously evaluated.

Allowing the PAL doors to be open during fuel movement and core alterations does increase the consequences of a fuel handling accident in the containment from essentially no offsite dose release to an estimated release of 65.6 rem to the thyroid and 0.28 rem to the whole body. However, the calculated offsite dose release is lower than the case analyzed in the FSAR [Final Safety Analysis Report] for an accident in the Spent Fuel Pool, with no filtration of the resulting release. In addition, the calculated doses are larger than the expected doses because the calculation does not incorporate the closing of the PAL door after the containment is evacuated. Closing the airlock door within 15 minutes results in a calculated offsite dose of 8.2 rem to the thyroid and 0.025 rem whole body. The projected dose to control room operators was reviewed and the projected dose remained below SRP acceptance limits as long as control room emergency ventilation was established within 7 minutes. It was assumed the individual assigned to close the airlock doors remained stationed at the airlock for 15 minutes. A best estimate dose analysis indicated this individual could be expected to receive 5.6 rem to the thyroid and 0.15 rem whole body. The proposed change will significantly reduce the dose to other workers in the containment in the event of a fuel handling accident by speeding the containment evacuation process. The proposed change will also significantly decrease the wear on the PAL doors and, consequently, increase the availability of the PAL doors in the event of an accident.

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

2. The proposed change to the Technical Specifications does not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed change affects a previously evaluated accident, e.g., a fuel handling accident. It does not represent a significant change in the configuration or operation of the plant and, therefore, does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. The proposed change to the Technical Specifications does not involve a significant reduction in a margin of safety. The margin of safety as defined by 10 CFR Part 100 for a fission product release is 300 rem thyroid and 25 rem whole body for an individual exposed at the site boundary for two hours. The analysis shows values that are well below the acceptance limits. In fact, the margin remains essentially the same as previously evaluated by the NRC. There is no increase in calculated offsite dose resulting from a fuel handling accident. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based upon the preceding information, it has been determined that the proposed Technical Specifications addition does not involve a significant hazards consideration as defined by 10 CFR 50.92.

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: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia 30830.

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308.

NRC Project Director: Herbert N. Berkow.

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: May 12, 1995.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) to support a one-time exemption from the requirement of Section III.D.1(a) of 10 CFR Part 50, Appendix J, and any other future Appendix J exemptions that may be approved by the NRC for Vogtle, Unit 1. Specifically, the TS change would insert the words ''Except as modified by NRC approved exemptions'' at the beginning of the first sentence of TS Surveillance Requirement 4.6.1.2.

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

1. The change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change does not involve a change to structures, systems, or components which would affect the probability of an accident previously evaluated in the Vogtle Electric Generating Plant (VEGP) Final Safety Analysis Report (FSAR). The change only provides a mechanism for implementing exemptions to 10 CFR 50, Appendix J containment leak rate testing criteria which have been approved by the NRC.

2. The proposed change will not create the possibility of a new or different kind of accident from any accident previously analyzed. The amendment would not change the design, configuration, or method of plant operation. It only allows exemption to specific 10 CFR 50, Appendix J criteria as previously approved by the NRC.

3. Operation of VEGP, Unit 1 in accordance with the proposed change will not involve a significant reduction in the margin of safety. The proposed change would not, in itself, change a safety limit, an LCO, or a surveillance requirement on equipment required for plant operation. Before the change could be used an exemption to 10 CFR 50, Appendix J would have to be evaluated and approved by the NRC. The change only provides a way to implement NRC approved exemptions without violating the Technical Specifications.

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

Local Public Document Room location: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia 30830.

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

Date of amendment request: June 1, 1995.

Description of amendment request: The proposed license amendment would revise the Technical Specifications (T.S.) for Three Mile Island Nuclear Station, Unit 1 (TMI–1) to delete the remaining portions of the TMI–1 Radiological Effluent Technical Specifications (RETS) and relocate them in accordance with the guidance contained in the Generic Letter 89–01 (GL 89–01) and NUREG–1430. The proposed change would also modify the Radiation Monitoring Systems surveillance requirements to specify only those radiation monitors that have Limiting Conditions for Operation (LCO), and revise some of the calibration frequencies.

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

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. The proposed amendment allows relocation of the remaining RETS to the ODCM [Offsite Dose Calculation Manual] according to the guidance contained in GL 89–01 and NUREG–1430. This proposal simplifies the RETS, meets the regulatory requirements for radioactive effluent controls and radiological environmental monitoring, and is provided as a line-item improvement of the T.S.

In addition, this proposed amendment specifies surveillance requirements only for those radiation monitors that have an LCO or specified operability requirements. The radiation monitors that are currently included in the T.S. surveillance program but have no associated LCO or specified operability requirement will be placed in the PM [preventive maintenance] program.

Finally, the proposed amendment extends the interval between successive calibration surveillances for those radiation monitors evaluated herein. This change does not involve any change to the actual surveillance requirements, nor does it involve any change to the limits or restrictions on plant operations. The reliability of systems and components relied upon to prevent or mitigate the consequences of accidents previously evaluated is not degraded beyond that obtained from the currently defined quarterly interval. Assurance of system and equipment availability is maintained.

This change does not involve any change to system or equipment configuration. Therefore, this change does not significantly increase the probability of occurrence or the consequences of an accident previously evaluated.

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

The proposal in part relocates procedural details, currently included in the T.S., on radioactive effluents to the ODCM. Future changes to these procedural details in the ODCM will be handled under the administrative controls for changes to the ODCM.

In addition, this proposed amendment specifies surveillance requirements only for those radiation monitors that have an LCO or specified operability requirements. The radiation monitors that are currently included in the T.S. surveillance program but have no associated LCO or specified operability requirement will be placed in the PM program.

Finally, the proposed amendment extends the interval between successive calibration surveillances for those radiation monitors evaluated herein. This change does not involve any change to the actual surveillance requirements, nor does it involve any change to the limits and restrictions on plant operations. This change does not involve any change to system or equipment configuration.

Therefore, this change is unrelated to the possibility of creating a new or different kind of accident from any previously evaluated.

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

The procedural details being relocated to the ODCM are consistent with the guidance provided in GL 89–01 and NUREG–1430.

In addition, this proposed amendment specifies surveillance requirements only for those radiation monitors that have an LCO or specified operability requirements. The radiation monitors that are currently included in the T.S. surveillance program but have no associated LCO or specified operability requirement will be placed in the PM program.

Finally, the proposed amendment extends the interval between successive calibration surveillances for those radiation monitors evaluated herein. This change does not involve any change to the actual surveillance requirements, nor does it involve any change to the limits and restrictions on plant operations. The reliability of the radiation monitors is not significantly degraded beyond that obtained from the currently defined surveillance interval. Assurance of system availability is maintained.

Therefore, it is concluded that operation of the facility in accordance with 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 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: Law/Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

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

NRC Project Director: Phillip F. McKee.

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

Date of amendment request: May 30, 1995.

Description of amendment request: The proposed amendment would revise the technical specifications (TS) to increase the surveillance test period for the containment integrated leak rate test (ILRT) from 40 plus or minus 10 months to every 10 years based on past performance. The change would also require testing on a more frequent basis if any test failures were to occur and to return to the 10 year period with subsequent performance improvements.

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 change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that contribute to initiation of any accidents previously evaluated. Thus, the proposed change cannot increase the probability of any accident previously evaluated. The proposed change potentially affects the leak tight integrity of the containment structure designed to mitigate the consequences of a loss of coolant accident (LOCA). The function of the containment is to maintain functional integrity during and following the peak transient pressures and temperatures which result from any loss of coolant accident (LOCA). The containment is designed to limit fission product leakage following the design basis LOCA and analyses demonstrate that these offsite doses are less than those allowed under 10CFR100 design limits of 15 psig and 185 °F. Because the proposed change does not alter the plant design, only the frequency of measuring containment leakage, the proposed change does not directly result in an increase in containment leakage. However, decreasing the test frequency can increase the probability that a large increase in containment leakage could go undetected for an extended period of time. These leakage paths include potential cracks in the containment structure and various penetrations through the containment structure. Based upon the results of the structural integrity test conducted as part of the preoperational or preservice test program and the periodic containment and drywell structural integrity surveillance tests, additional cracking of the containment is not expected during the remaining life to the plant. Ventilation and piping penetrations are designed with two isolation valves in series with one valve in the drywell and another either outside primary containment or in the wetwell. High energy lines that extend into the wetwell, such as the Main Steam and Feedwater lines, are encapsulated by guard pipes to direct energy to the drywell in case of a piping rupture.

Electrical penetrations are sealed with a high strength/density material that will prevent leakage as well as provide radiation shielding. The TS ILRT acceptance criterion of 0.75 L_a [maximum allowable leakage rate at the calculated maximum accident pressure, P_a] provides margin for degradation. Containment performance data to date suggests that containment degradation, even during a ten (10) year interval between tests, will not exceed this margin.

Based on the above, EOI [Entergy Operations, Inc.] has concluded that the proposed change will not result in a significant increase in the probability or consequences of any accident previously evaluated.

(2) The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to initiation of any accidents. This change involves the reduction in the Integrated Leak Rate Test frequency. The method of performing the test is not changed. No new accident modes are created by extending the testing intervals. No safety-related equipment or safety functions are altered as a result of this change. Extending the test frequency has no influence on, nor does it contribute to, the possibility of a new or different kind of accident or malfunction from those previously analyzed. Based upon the above, EOI has concluded that the proposed change will not create the possibility or a new or different kind of accident previously evaluated.

(3) The proposed change only affects the frequency of measuring containment leakage and does not change the leakage rate limit. However, the proposed change can increase the probability that a large increase in containment leakage could go undetected for an extended period of time. Operational experience has shown that the leak tightness of the containment has been maintained significantly below the allowable leakage limit. In fact, an analysis was conducted to determine the potential risk to the public from the proposed change. Based on this analysis, under several different accident scenarios, the risk of radioactivity release from containment was found to be negligible.

The margin of safety that has the potential of being impacted by the proposed change involves the offsite dose consequences of postulated accidents which are directly related to containment leakage rate. The containment isolation system is designed to limit leakage to L_a which is defined by the RBS Technical Specifications to be 0.26 percent by weight of the containment air per 24 hours at 7.6 psig (P_a). The limitation on containment leakage rate is designed to ensure that total leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure (P_a) or 7.6 psig.

To provide additional conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic Integrated Leak Rate Test and to less than or equal to 0.60 L_a (total combined leakage) for Type B and C leak rate tests. This is done to account for the possible degradation of the containment leakage barriers between tests. These acceptance criteria ensure that an acceptable margin of safety is being maintained and will not be altered by the proposed change. The preservation of this margin will continue to provide for potential degradation of the leakage barriers between tests. RBS [River Bend Station] presently has on docket with the staff a submittal (reference RBG–41133, Rev. 1 to LAR 93–14 dated January 18, 1995) that allows the acceptance criteria, between required leakage rate tests, to be less than or equal to 1.0 L_a since at less than or equal to 1.0 L_a, the offsite does consequences are bounded by the assumptions of safety analysis.

No change in the method of testing is being proposed. The Type A test will continue to be done at full pressure (P_a) or greater. Primary containment penetrations which require Type B or C leak tests will be performed in the same manner as before. Other programs are in place to ensure that proper maintenance and repairs are performed during the service life of the primary containment and systems and components penetrating the primary containment.

No change in the RBS allowable leakage rate is being proposed. These conservative leakage rates ensure that the containment leakage remains low. As a result, EOI has concluded that the proposed change will not result in 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: Government Documents Department, Louisiana State University, Baton Rouge, Louisiana 70803.

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

NRC Project Director: William D. Beckner.

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

Date of amendment request: May 30, 1995.

Description of amendment request: The proposed amendment would revise the technical specifications (TS) to increase the time period for drywell leakage tests from eighteen months to five years based on performance. The new surveillance requirements would also reduce the time period if any failures occur and limit subsequent periods until drywell leakage test performance again improves.

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 change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that contribute to initiation of any accidents previously evaluated. Thus, the proposed change cannot increase the probability of any accident previously evaluated.

The proposed change potentially affects the leak tight integrity of the drywell, a structure used to mitigate the consequences of a loss of coolant accident (LOCA). The function of the drywell is to channel the steam released from a LOCA through the suppression pool, limiting the amount of steam released to the primary containment atmosphere. This limits the containment pressurizations due to the LOCA. The leakage of the drywell is limited to ensure that the primary containment does not exceed its design limits of 185°F and 15 psig. Because the proposed change does not alter the plant design, only the frequency of measuring the drywell leakage, the proposed change does not directly result in an increase in drywell leakage. However, decreasing the test frequency can increase the probability that a large increase in drywell bypass leakage could go undetected for an extended period of time. There are several potential sources of steam bypass leakage paths. These include potential cracks in the drywell concrete structure and various penetrations through the drywell structure. Based upon the results of the structural integrity test conducted as part of the preoperational or preservice test program, additional cracking of the drywell is not expected during the remaining life of the plant. Ventilation and piping penetrations are designed with two isolation valves in series with one valve in the drywell and another either outside primary containment or in the wetwell. High energy lines that extend into the wetwell, such as the Main Steam line and Feedwater lines, are encapsulated by guard pipe to direct energy to the drywell in case of a piping rupture. Electrical penetrations are sealed with a high strength/density material that will prevent leakage as well a provide radiation shielding. The TS DBLRT [Drywell Bypass Leakage Rate Tests] acceptance criterion of 10% of the design bypass leakage area parameter provides margin for degradation. Drywell performance data to date suggests that drywell degradation, even during a five year interval between tests, will not exceed this margin. RBS presently has on docket with the staff a submittal (reference EOI letter RBG–41133, Rev. 1 to LAR 93–14 dated January 18, 1995) that allows the acceptance criteria, between required leakage rate tests, to be (bypass leakage area parameter) since at (bypass leakage area parameter) the containment temperature and pressurization response are bounded by the assumptions of the safety analysis.

Based on the above, EOI has concluded that the proposed change will not result in a significant increase in the consequences of any accident previously evaluated.

(2) The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to initiation of any accidents. Thus, the proposed change cannot create the possibility of an accident not previously evaluated.

(3) The proposed change only affects the frequency of measuring the drywell bypass leakage rate and does not change the bypass leakage limit for the drywell. However, the proposed change can increase the probability that a large increase in drywell bypass leakage could go undetected for an extended period of time. Operational experience has shown that the leak tightness of the drywell has been maintained significantly below the allowable leakage limits. In fact, an analysis was conducted to determine the potential risk to the public from the proposed change. Based on this analysis, under several different accident scenarios, the risk of radioactivity release from containment was found to be negligible.

As a result, EOI has concluded that the proposed change will not result in 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: Government Documents Department, Louisiana State University, Baton Rouge, Louisiana 70803.

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

NRC Project Director: William D. Beckner.

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

Date of amendment requests: May 25, 1995 (AEP:NRC:107IT).

Description of amendment requests: The proposed amendments would implement a cycle- and burnup-dependent peaking factor penalty to the allowable power level. The Technical Specifications would be changed to refer to the Core Operating Limits Report for this burnup-dependent penalty.

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:

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) 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, or

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

Criterion 1

The proposed changes will not involve a significant increase in the probability of an accident previously evaluated because the changes will not result in a change to any of the process variables that might initiate an accident. There are no physical changes to the plant associated with this T/S change. The consequences of an accident previously evaluated will not be increased because the changes increase the penalty applied to F_Q when it is measured to be increasing. F_Q and allowable power level (APL) T/S surveillance requirements are not being changed. Furthermore, allowing a cycle and burnup dependent F_Q penalty to be located in the COLR was accepted by the NRC in a [November 26, 1993] safety evaluation on WCAP–10216–P, Rev. 1 [''Relaxation of Constant Axial Offset Control- F_Q Surveillance Technical Specification''].

Criterion 2

The proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated because the changes will involve no physical changes to the plant nor any changes in plant operations. Furthermore, the F_Q and APL T/S surveillance requirements are not being changed, and the change to the F_Q penalty is conservative.

Criterion 3

The proposed amendment[s] will not involve a significant reduction in a margin of safety. When the increased F_Q penalty is applied, it reduces the allowable power level, thus increasing 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 requests involve no significant hazards consideration.

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

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

NRC Project Director: Cynthia A. Carpenter, Acting.

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

Date of amendment requests: May 25, 1995 (AEP:NRC:1124B).

Description of amendment requests: The proposed amendments would modify the Technical Specifications (TS) to allow fuel reconstitution. The proposed change is a TS line item improvement per NRC Generic Letter 90–02, supplement 1, ''Alternative Requirements for Fuel Assemblies in the Design Features Section of Technical Specifications.''

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

Per 10 CFR 50.92, a proposed change does not involve significant hazards consideration if the change does not:

1. Involve a significant increase in the probability or consequence of an accident previously evaluated,
2. Create the possibility of a new or different kind of accident from any accident previously evaluated, or
3. Involve a significant reduction in a margin of safety.

Criterion 1

The proposed changes only modify the T/Ss such that reconstitution is recognized as acceptable under very limited circumstances. Reconstitution is limited to substitution of zirconium alloy or stainless steel filler rods, and must be in accordance with approved applications of fuel rod configurations. Although these changes permit reconstitution to occur without the need for a specific T/S change, an approved methodology is required prior to its application. Since the changes will allow substitution of filler rods for leaking or potentially leaking rods, the changes may actually reduce the radiological consequences of an accident. It is noted that the specific changes requested in this letter have previously been found acceptable by the NRC in GL 90–02 supplement 1. For these reasons, we conclude that the changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2

The proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated because they will only affect the assembly configuration and can only be implemented in accordance with an NRC-approved methodology. The other aspects of plant design, operation limitations, and responses to events will remain unchanged. It is noted that the changes have previously been determined acceptable by the NRC in GL 90–02 supplement 1.

Criterion 3

The proposed amendment will not involve a significant reduction in a margin of safety because the changes can only be implemented in accordance with an NRC-approved methodology. It is noted that the changes have previously been determined acceptable by the NRC in GL 90–02 supplement 1.

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

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

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

NRC Project Director: Cynthia A. Carpenter, Acting.

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

Date of amendment requests: May 25, 1995 (AEP:NRC:1200B).

Description of amendment requests: The proposed amendments would modify the Technical Specifications to change the surveillance frequency of the manual actuation function for main steam line isolation. This change is consistent with the testing requirements for associated valves as specified in the American Society of Mechanical Engineers (ASME) Code Section XI inservice testing program at Cook.

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:

Per 10 CFR 50.92, a proposed change does not involve significant hazards consideration if the change does not:

1. Involve a significant increase in the probability or consequence of an accident previously evaluated,
2. Create the possibility of a new or different kind of accident from any accident previously evaluated, or
3. Involve a significant reduction in a margin of safety.

Criterion 1

This change will reduce the frequency of the surveillance testing on the MSIV [main steamline isolation valve] manual actuation circuitry from monthly to quarterly. Because of the risks involved in testing the dump valves, the reduction in test frequency may reduce the probability of an accidental unit trip and valve seat failure due to repeated cycling. Our review of the surveillance test history has shown that the system is highly reliable, and gives us confidence that the change in test frequency will not endanger public health and safety. Furthermore, the change to a quarterly surveillance interval is consistent with the testing performed for the dump valves per ASME Section XI. For these reasons, it is our belief that the proposed changes do not involve a significant increase in the probability or consequences of a previously evaluated accident.

Criterion 2

The changes will not introduce any new modes of plant operation, nor will any physical changes to the plant be required. Thus, the changes should not create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated.

Criterion 3

This change will reduce the frequency of the surveillance testing on the MSIV manual actuation circuitry from monthly to quarterly. Our review of the surveillance test history has shown that the system is highly reliable, and gives us confidence that the change in test frequency will not endanger public health and safety. Furthermore, the change to quarterly surveillance is consistent with the testing performed for the dump valves per ASME Section XI. For these reasons, it is our belief that the proposed changes do not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Cynthia A. Carpenter, Acting.

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

Date of amendment requests: May 26, 1995 (AEP:NRC:1210).

Description of amendment requests: The proposed amendments would modify the Reactor Trip System Instrumentation and Engineered Safety Feature Actuation System Instrumentation sections of the Technical Specifications (TS) to relocate the tables of response time limits to the Updated Final Safety Analysis Report (UFSAR). These changes are a line item improvement of the TS in accordance with NRC Generic Letter 93–08, ''Relocation of Technical Specification Tables of Instrument Response Time 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:

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) 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, or

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

Criterion 1

The proposed changes will not involve a significant increase in the probability of an accident previously evaluated because the changes will not result in a change to any of the process variables that might initiate an accident. There are no physical changes to the plant associated with the T/S change. The consequences of an accident previously evaluated will not be increased because the changes simply allow relocation of response time limits to the UFSAR. Time response testing will continue to be required by the T/Ss. Any changes to the response time values will be made in accordance with the requirements of 10 CFR 50.59. It is noted that these T/S changes have previously been determined acceptable by the NRC in GL 93–08.

Criterion 2

The proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated because the changes will involve no physical changes to the plant nor any changes in plant operations. Time response testing will continue to be required by the T/Ss. Any changes to the time response values will be made in accordance with the requirements of 10 CFR 50.59. It is noted that these changes have previously been determined acceptable by the NRC in GL 93–08.

Criterion 3

The proposed amendment will not involve a significant reduction in a margin of safety because time response testing will continue to be required by the T/Ss. Any changes to the response time values will be made in accordance with the requirements of 10 CFR 50.59. It is noted that these changes have previously been determined acceptable by the NRC in GL 93–08.

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

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

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

NRC Project Director: Cynthia A. Carpenter, Acting.

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

Date of amendment request: May 30, 1995.

Description of amendment request: The proposed amendment would change the upper limit for the moderator temperature coefficient (MTC) for certain operating conditions. Specifically, the upper limit specified in Technical Specification 3.1.1.3 for the MTC would be changed to +0.5 x 10^{-4} delta k/k/°F for all rods out at the beginning of cycle for power levels up to 70% rated thermal power with a linear ramp to 0 delta k/k/°F at 100% rated thermal power. The currently specified upper limit for all operating conditions is 0 delta k/k/°F.

A paragraph would be added to the Basis to Technical Specification 3.1.1.3 providing a commitment to comply with the ATWS Rule and the basis for the Rule by assuring ATWS core damage frequency will remain below the Commission established target of 1.0 x 10^{-5} per reactor year. The commitment would be implemented by determining a more restrictive, cycle-specific upper MTC limit and placing it in the Core Operating Limits Report (COLR).

Additionally, a reference for the analytical method used to determine the cycle-specific MTC upper limit would be added to TS 6.8.1.6.b.

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

A. The changes do not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)). The proposed changes do not affect the manner by which the facility is operated and do not change any facility design feature or equipment which influences the initiation of an accident, therefore, there is no change in the probability of any accident previously analyzed. Each accident or transient, with the exception of the Anticipated Transient Without SCRAM (ATWS), has been analyzed for the proposed changes and has been approved previously by the Commission with the issuance of Amendment 33 (December 6, 1994) to the Facility Operating License. The proposed cycle-specific MTC to be included in the COLR will assure that the consequences of an ATWS will remain bounded by the analysis previously documented. Therefore, the consequences of previously evaluated accidents, including ATWS, will not be significantly increased by the proposed changes.

B. The changes do not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because the changes proposed merely involve changes in the upper limits of MTC imposed by the Technical Specifications and COLR. No changes are made to the design or manner of operation of any structure, system or component and no new failure mechanisms are introduced.

C. The changes do not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)). The analyses of each accident or transient previously presented to support the issuance of Amendment 33 were performed using the proposed upper MTC limit, and the results demonstrated that the acceptance criteria specified for each event are met. The cycle-specific MTC limit in the COLR will be adjusted to assure that the acceptance criteria for a postulated ATWS event are met thereby preserving the margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: Thomas Dignan, Esquire, Ropes & Gray, One International Place, Boston MA 02110–2624.

NRC Project Director: Phillip F. McKee.

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

Date of amendment request: May 31, 1995.

Description of amendment request: The amendment would provide additional restrictions on the operation of the component cooling water (CCW) system heat exchangers to ensure that the CCW system temperature is maintained within its analyzed design basis.

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.

In preparation for, and in response to a service water system operational performance self assessment, the heat loads in the Component Cooling Water (CCW) system were reevaluated to determine the peak temperatures on the system and components cooled by the CCW system. It was determined that if all of the containment coolers were operating, the return temperature of the CCW system could exceed the 120°F stated in the Updated Safety Analysis Report (USAR) as the maximum temperature of the system.

During a Large Break Loss of Coolant Accident (LBLOCA) or a Main Steam Line Break Inside Containment (MSLB/IC), the containment air cooling units and containment air cooling and filtering units will automatically start to remove heat from the containment atmosphere. The heat sink for the containment air coolers is the CCW system. The heat removed from the containment atmosphere is transferred to the Raw Water (RW) system via the component cooling heat exchangers AC–1A, B, C, and D. The heat is then ultimately rejected to the Missouri River by the RW system.

Calculations indicate that the CCW return temperature (i.e., mixed exit temperature) from the component cooling heat exchangers could exceed 160°F after a LBLOCA or MSLB/IC with the present TS minimum requirements for the heat exchangers. Further evaluation indicated that the CCW system (and components cooled by CCW) could withstand temperatures above the 120°F temperature stated in the USAR, but a return temperature above 158°F would require additional evaluation of thermal-induced stresses on the CCW return side pipe supports. In order to maintain the peak CCW return temperature to less than or equal to 158°F, additional restrictions must be placed on the number of component cooling heat exchangers required to be operable.

The current minimum requirements for component cooling heat exchangers are contained in Technical Specification (TS) 2.3, ''Emergency Core Cooling System,'' and require that three of the four heat exchangers be operable when the plant is in operating Modes 1 and 2. Analyses show that three in service heat exchangers will maintain the CCW temperatures in an analyzed range following a DBA. In order to ensure that three heat exchangers are available, in conjunction with an assumed single failure, four are required to be operable. The proposed change would place additional restrictions on the operation of the CCW heat exchangers by requiring four heat exchangers to be operable in Modes 1 and 2, and if only three are operable then provide 14 days to restore the system to four operable heat exchangers.

The proposed change does not involve a significant increase in the probability of an accident previously evaluated. The proposed change does not impact systems, structures, or components that are initiators of any analyzed accidents.

The proposed change does not involve a significant increase in the consequences of an accident previously evaluated. The proposed change ensures that the CCW system and safety-related components cooled by the CCW will perform their safety functions in response to previously evaluated accidents. The proposed change was evaluated utilizing the probabilistic risk analysis model of the FCS Individual Plant Examination. The IPE concluded that the routine testing and maintenance activities, for the RW and CCW systems (e.g., inoperability of components for testing and maintenance) are not significant contributors to severe accident risk.

Therefore, the proposed change would not increase the probability or consequences of an accident previously evaluated.

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

The proposed change does not create an initiator for a new or different kind of accident from those previously evaluated. The proposed change places additional restrictions on the operation of equipment to ensure that the CCW system and safety-related components cooled by the CCW will perform their safety functions. The additional restrictions were evaluated in combination with existing allowances on RW and CCW pump inoperability, to confirm that the peak CCW return temperature would be in an analyzed range, and will not adversely impact the operability of the CCW system or safety-related components cooled by CCW. These restrictions are valid up to and including a river temperature of 90°F, which is the upper bound currently cited in the USAR.

Various single active failures were postulated to determine the most limiting failure in conjunction with the maximum heat load from the containment air coolers. It was determined that with the river temperature less than 70 °F, a single failure of a RW valve to open on a component cooling heat exchanger would not raise the CCW return temperature to an unanalyzed level, but with the river temperature greater than or equal to 70 °F, the CCW return temperature could be at an unanalyzed level. Therefore, it is proposed that when the river temperature is greater than or equal to 70 °F four heat exchangers have RW in service (i.e., RW valves open). Having RW in service eliminates the potential failure of a RW valve to auto-open as a credible single active failure.

The proposed change ensures that the CCW system and safety-related components cooled by the CCW will perform their safety functions. 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 proposed change provides additional restrictions on the CCW system and ensures that the CCW system will perform its design safety function. These additional restrictions ensure that the CCW system will be capable of removing the maximum heat load from the containment cooling system following a DBA and thereby ensures that the containment pressure remains below its limit as assumed in the USAR. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: James R. Curtiss, Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: William H. Bateman.

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

Date of amendment request: May 5, 1995.

Description of amendment request: This amendment would remove from the Susquehanna Steam Electric Station Unit 2 Technical Specifications, the listing of three residual heat removal (RHR) system valves in Table 3.6.3–1, ''Primary Containment Isolation Valves'' These valves are no longer needed to support the steam condensing mode of the RHR system and are being removed from the plant during the Unit 2 seventh refueling and inspection outage in September of this year.

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.

With the prior deletion of the steam condensing mode of RHR and the isolation of the high and low pressure interfaces, the three pressure relief valves that are being removed from the plant have no active function. Their passive function of maintaining system or containment integrity will be fulfilled by blind flanges on equilvent. Also, the RHR and RCIC piping are provided with overpressure protection from other pressure relief valves. Therefore, the removal of these pressure relief valves 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 pressure relief valves that are being removed had two primary functions. First, they provided overpressure protection for the RHR and RCIC piping during the steam condensing mode of RHR. Since the steam condensing mode has been deleted from the plant, these valves no longer have that function. Also, overpressure protection of the RHR and RCIC piping is provided by other existing pressure relief valves. Second, these valves maintained system or containment integrity. When the pressure relief valves are removed from the plant, they will be replaced with blind flanges or equivalent that will maintain system or containment integrity. Therefore, the removal of the three pressure relief valves 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.

Since the steam condensing mode of RHR has been eliminated, the three pressure relief valves have no active function. Their passive function of maintaining system or containment integrity will be fulfilled by blind flanges or equivalent. Also, overpressure protection of RHR and RCIC piping is provided by other existing pressure relief valves. Therefore, the removal of the three pressure relief valves does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: May 19, 1995.

Description of amendment request: The proposed Technical Specifications (TS) change would revise TS Table 3.3.3–3, ''Emergency Core Cooling System Response Times'' to reflect the value of 60 seconds for the High Pressure Coolant Injection system response time instead of 30 seconds as currently specified.

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

The proposed TS change will increase the High Pressure Coolant Injection (HPCI) system response time from 30 seconds to 60 seconds. The proposed TS change does not involve any physical change in the plant configuration which may cause an accident, or affect safety-related equipment performance or cause its failure. There is no increase in the consequences of an accident, because the HPCI response time increase does not affect the licensing basis Peak Cladding Temperature (PCT), which remains below the regulatory limit of 2200 °F.

The Loss of Feedwater Flow (LOFW) event was evaluated for being potentially affected by the increased HPCI system response time. The HPCI system is one of the systems which provides reactor vessel water makeup inventory, and is initiated automatically on a low reactor water level (Level 2) signal. The LOFW analysis shows that Level 1 is not reached and that the top of the active fuel will remain covered throughout the event. Therefore, adequate core cooling will be maintained and no fuel damage will result. The probability of fuel failure will not be increased by this proposed TS change.

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

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

The proposed TS change will increase the High Pressure Coolant Injection (HPCI) system response time from 30 seconds to 60 seconds. This proposed change is bounded by the current Emergency Core Cooling System (ECCS)—Loss-of-Coolant Accident (LOCA) analysis for Limerick Generating Station (LGS) Units 1 and 2. The change in HPCI system response time does not involve any physical modifications to the plant systems or equipment, nor does it introduce a new operational/failure mode, which might cause a different type of accident. In case of a Loss of Feedwater Flow (LOFW) event, the HPCI system will operate as designed, maintaining adequate core cooling.

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 TS change does not involve a significant reduction in a margin of safety.

The following TS Bases were reviewed for potential reduction in the margin of safety:

3/4.5 Emergency Core Cooling System
2.1.4 Reactor Vessel Water Level

The TS Bases do not discuss the High Pressure Coolant Injection (HPCI) system start time. The margin of safety, as defined in the TS Bases, will remain the same. The proposed TS change is in accordance with the current licensing basis Emergency Core Cooling System (ECCS)—Loss of Coolant Accident (LOCA) analysis for LGS Units 1 and 2, and does not impact any safety limits of the plant. The HPCI system will operate as designed during the LOFW event, maintaining adequate core cooling.

Therefore, the proposed TS change does 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: Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

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

NRC Project Director: John F. Stolz.

Notice of Issuance of Amendments to Facility Operating Licenses

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

Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the **Federal Register** as indicated.

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

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

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of application for amendments: December 7, 1994.

Brief description of amendments: These amendments revise the Bases of TS 3/4.7.5, ''Ultimate Heat Sink'' (UHS), to describe the UHS as containing a 26-day supply of cooling water, instead of a 27-day supply. In addition, the reference to Regulatory Guide 1.27 in the bases of this TS would be revised to reference the January 1976 revision rather than the March 1974 revision.

Date of issuance: June 14, 1995.

Effective date: June 14, 1995.

Amendment Nos.: Unit 1—Amendment No. 93; Unit 2—Amendment No. 81; Unit 3—Amendment No. 64.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the associated Bases of the Technical Specifications.

Date of initial notice in **Federal Register**: March 1, 1995 (60 FR 11127) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 14, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: February 9, 1995.

Brief description of amendment: This amendment revises the reactor high water level trip level setting for the Group 1 isolation. The change will allow an increase to the main steam isolation valve high water level isolation setpoint.

Date of issuance: June 15, 1995.

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

Amendment No.: 164.

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

Date of initial notice in **Federal Register**: March 15, 1995 (60 FR 14017) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 15, 1995.

No significant hazards consideration comments received: No.

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

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

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

Date of application for amendments: May 20, 1994, as revised on February 2, 1995, and supplemented December 2, 1994, and March 14, 1995.

Brief description of amendments: The amendments revised the Technical Specifications (TS) as they apply to Byron, Unit 1, and Braidwood, Unit 1, to incorporate an alternative repair criteria for defects found in the portion of the expanded steam generator tubes within the tubesheet.

Date of issuance: June 22, 1995.

Effective date: June 22, 1995.

Amendment Nos.: 72, 72, 63, and 63.

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

Date of initial notice in **Federal Register**: July 6, 1994 (59 FR 34659) and March 29, 1995 (60 FR 16184). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 22, 1995.

No significant hazards consideration comments received: No.

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

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 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

Date of application for amendments: September 15, 1992, as supplemented April 21, 1995.

Brief description of amendments: This application upgrades the current custom Technical Specifications (TS) for Dresden and Quad Cities to the Standard Technical Specifications contained in NUREG–0123, ''Standard Technical Specification General Electric Plants BWR/4.'' This application upgrades only Sections 2.0 (Safety Limits and Limiting Safety System Settings), 3/4.11 (Power Distribution Limits), and 3/4.12 (Special Test Exceptions).

Date of issuance: June 13, 1995.

Effective date: Immediately, to be implemented no later than December 31, 1995, for Dresden Station and June 30, 1996, for Quad Cities Station.

Amendment Nos.: 134, 128, 155, and 151.

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

Date of initial notice in **Federal Register**: May 10, 1995 (60 FR 24906) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 13, 1995.

No significant hazards consideration comments received: No.

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

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

Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: December 15, 1993, as supplemented April 21, 1995.

Brief description of amendments: These amendments upgrade the current custom Technical Specifications (TS) for Dresden and Quad Cities to the Standard Technical Specifications contained in NUREG–0123, ''Standard Technical Specifications General Electric Plants BWR/4.'' These amendments upgrade only Section 5.0 (Design Features). The amendments include the relocation of some requirements from the TS to licensee-controlled documents.

Date of issuance: June 14, 1995.

Effective date: Immediately, to be implemented no later than December 31, 1995, for Dresden Station and June 30, 1996, for Quad Cities Station.

Amendment Nos.: 135, 129, 156, and 152

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

Date of initial notice in **Federal Register**: May 10, 1995 (60 FR 24909) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 14, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: December 13, 1994, as supplemented May 3, 1995.

Brief description of amendment: This amendment revises the Technical Specifications to add a high thermal performance (HTP) departure from nucleate boiling correlation to Safety Limit 2.1. The HTP correlation is used for HTP fuel loaded during recent fuel cycles.

Date of issuance: June 13, 1995.

Effective date: June 13, 1995.

Amendment No.: 168.

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

Date of initial notice in **Federal Register**: March 10, 1995 (60 FR 24910) The May 3, 1995, submittal provided clarifying information which was within the scope of the initial application and did not affect the staff's initial proposed no significant hazards considerations findings.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 13, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 17, 1993, as supplemented October 20, 1993, and May 23, 1995.

Brief description of amendments: These amendments revise the Appendix A technical specifications (TSs) for Unit 1 and Unit 2 by relocating the requirements of the radiological effluent technical specifications (RETS) and the solid radioactive wastes TSs from the Appendix A TSs to the offsite dose calculation manual (ODCM) or to the process control program (PCP) in accordance with the guidance provided in NRC Generic Letter 89–01 and NRC Report NUREG–1301. Programmatic controls are also being incorporated into the Administrative Controls section of the TSs. Additionally, editorial and definition changes are being made to facilitate the relocation of these requirements.

Date of issuance: June 12, 1995.

Effective date: June 12, 1995.

Amendment Nos.: 188 and 70.

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

Date of initial notice in **Federal Register**: August 4, 1993 (58 FR 41504). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 12, 1995.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: May 15, 1995, as supplemented by letters dated May 19 and June 7, 1995.

Brief description of amendment: The amendment was processed as an exigent amendment following issuance of a notice of enforcement discretion (NOED) by NRC letter dated May 17, 1995. The NOED and exigent technical specification (TS) amendment authorized the licensee to continue operating the reactor at power while the service water flow to the reactor building emergency coolers is less than the TS surveillance criteria.

Date of issuance: June 9, 1995.

Effective date: June 9, 1995.

Amendment No.: 182.

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

Public comments requested as to proposed no significant hazards consideration: Yes (60 FR 27144, dated May 22, 1995). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by June 21, 1995, but stated that any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendments, finding of exigent circumstances, and final determination of no significant hazards consideration is contained in a Safety Evaluation dated June 9, 1995.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: January 27, 1995.

Brief description of amendment: The amendment changed the Appendix A Technical Specifications by increasing the allowable maximum enrichment for the spent fuel pool and containment temporary storage rack from 4.1 to 4.9 weight percent U–235 when fuel assemblies contain fixed poisons.

Date of issuance: June 14, 1995.

Effective date: June 14, 1995.

Amendment No.: 108.

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

Date of initial notice in **Federal Register**: March 15, 1995 (60 FR 14021)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 14, 1995.

No significant hazards consideration comments received: No.

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

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: February 27, 1995.

Brief description of amendment: This amendment will modify surveillance requirement (SR) 4.9.8.1 and 4.9.8.2 to allow a reduction in the required minimum shutdown cooling flow rate under certain conditions during operational MODE 6. In addition, the format of the SR will be changed to clarify the intent of the stated surveillances.

Date of Issuance: June 14, 1995.

Effective Date: June 14, 1995.

Amendment No.: 76.

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

Date of initial notice in **Federal Register***:* March 29, 1995 (60 FR 16187) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 14, 1995.

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 Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of application for amendments: February 22, 1995.

Brief description of amendments: The proposed changes eliminate reference to an automatic containment air lock tester from technical specification 4.6.1.3. The automatic air lock tester is no longer being used.

Date of Issuance: June 22, 1995.

Effective Date: June 22, 1995.

Amendment Nos.: 137 and 77.

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

Date of initial notice in **Federal Register***:* March 29, 1995 (60 FR 16186) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 22, 1995.

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, Docket Nos. 50–250 and 50–251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Date of application for amendments: January 17, 1995.

Brief description of amendments: These amendments concern implementation of Florida Power and Light nuclear physics methodology for calculations of the core operating limits report parameters.

Date of issuance: June 9, 1995.

Effective date: June 9, 1995.

Amendment Nos. 174 and 168.

Facility Operating Licenses Nos. DPR–31 and DPR–41: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register***:* March 1, 1995 (60 FR 11133) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 9, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199.

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: October 3, 1994, as supplemented by letter dated March 1, 1995.

Brief description of amendments: The amendments revise Technical Specification 3/4.4.9, Pressure/Temperature Limits, and its associated Bases, to provide new reactor coolant system heatup and cooldown limitations and new power-operated relief valve setpoints for the low temperature overpressure protection system.

Date of issuance: June 8, 1995.

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

Amendment Nos.: 87 and 65.

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

Date of initial notice in **Federal Register***:* December 21, 1994 (59 FR 65814) The March 1, 1995, letter provided supporting technical data that did not change the scope of the October 1, 1994, application and initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 8, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Burke County Library, 412 Fourth Street, Waynesboro, Georgia 30830.

GPU Nuclear Corporation, Docket No. 50–320, Three Mile Island Nuclear Station, Unit No. 2, (TMI–2), Dauphin County, Pennsylvania

Date of application for amendment: October 9, 1991.

Brief description of amendment: This amendment extends the expiration date of the license from November 9, 2009 to April 19, 2014.

Date of issuance: June 21, 1995.

Effective date: June 21, 1995.

Amendment No.: 49.

Possession-Only License No. DPR–73: The amendment extends the license expiration date.

Date of initial notice in **Federal Register**: August 3, 1994 (59 FR 39591). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 21, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105.

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

Date of amendment request: February 22, 1994, as supplemented May 19, 1995.

Brief description of amendment: The amendment revised Technical Specifications 3.6.1.5, ''Main Steam—Positive Leakage Control System,'' and 3.6.1.10, ''Penetration Valve Leakage Control System,'' to add an allowed outage time of 7 days with both trains of each system inoperable. In addition, the allowed outage time for one train of the Penetration Valve Leakage Control System inoperable is increased from 7 days to 10 days.

Date of issuance: June 19, 1995.

Effective date: June 19, 1995.

Amendment No.: 80.

Facility Operating License No. NPF–47. The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register***:* March 10, 1994 (59 FR 11331) The additional information contained in the supplemental letter dated May 19, 1995, was clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 19, 1995.

No significant hazards consideration comments received. No.

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

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

Date of application for amendments: November 17, 1994 as supplemented March 30, 1995.

Brief description of amendments: The amendments change equipment designations, instrument range descriptions, instrument setpoints and surveillance requirements in the Peach Bottom Technical Specifications to reflect planned modifications to the main stack and vent stack radiation monitoring systems.

Date of issuance: June 13, 1995.

Effective date: June 13, 1995.

Amendments Nos.: 204 and 207.

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

Date of initial notice in **Federal Register***:* March 15, 1995 (60 FR 14027) The March 30, 1995, submittal provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 13, 1995.

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, Pennsylvania 17105.

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

Date of application for amendments: March 16, 1995.

Brief description of amendments: These amendments change the existing Technical Specification requirements for source range neutron monitoring equipment while in the refueling mode to requirements based on NUREG–1433, ''Standard Technical Specifications General Electric Plants, BWR/4.''

Date of issuance: June 13, 1995.

Effective date: June 13, 1995.

Amendments Nos.: 205 and 208.

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

Date of initial notice in **Federal Register***:* May 10, 1995 (60 FR 24913) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 13, 1995.

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, Pennsylvania 17105.

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: March 30, 1995, as supplemented by letter dated May 26, 1995.

Brief description of amendment: The proposed amendment revises Technical Specification Section 4.7.D.1.b(1) by adding a footnote to exempt the High Pressure Coolant Injection motor-operated valve MO–2–23–015 from quarterly stroke testing requirements until refueling outage 2RO11.

Date of issuance: June 13, 1995.

Effective date: June 13, 1995.

Amendment No.: 206.

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

Date of initial notice in **Federal Register***:* May 10, 1995 (60 FR 24912) The May 26, 1995, submittal provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 13, 1995.

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, Pennsylvania 17105.

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

Date of application for amendments: March 22, 1995.

Brief description of amendments: These amendments reduce the local leak rate test hold time specified in the Technical Specification Tables 3.7.2 through 3.7.4 from one hour to 20 minutes.

Date of issuance: June 19, 1995.

Effective date: June 19, 1995.

Amendments Nos.: 207 and 209.

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

Date of initial notice in **Federal Register***:* May 10, 1995 (60 FR 24913). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 19, 1995.

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, Pennsylvania 17105.

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

Date of application for amendments: October 28, 1994, as supplemented by letter dated April 18, 1995.

Brief description of amendments: These amendments delete, from the Technical Specifications, the surveillance and operability requirements for chlorine detection and the associated Bases as a result of the removal of bulk quantities of gaseous chlorine from the site.

Date of issuance: June 19, 1995.

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

Amendment Nos.: 147 and 117.

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

Date of initial notice in **Federal Register***:* December 21, 1994 (59 FR 65821). The April 18, 1995, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 19, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: August 31, 1994.

Brief description of amendments: This amendment revises the Technical Specifications to permit the operability requirement for the Feedwater/Main Turbine Trip System Actuation Instrumentation to be Operational Condition 1 greater than or equal to 25% Rated Thermal Power.

Date of issuance: June 13, 1995.

Effective date: June 13, 1995.

Amendment Nos. 91 and 55.

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

Date of initial notice in **Federal Register:** November 9, 1994 (59 FR 55884) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 13, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: August 23, 1994.

Brief description of amendments: Remove the 125/250 Vdc Class 1E Battery Load Cycle Table from the technical specifications (TS) and rephrase the surveillance requirements to be consistent with NUREG–1433, ''Standard Technical Specifications'', and correct Amendments 71 and 34, dated June 28, 1994, to change certain surveillance requirement intervals from 24 months to 18 months.

Date of issuance: June 19, 1995.

Effective date: June 19, 1995.

Amendment Nos. 92 and 56.

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

Date of initial notice in **Federal Register:** October 12, 1994 (59 FR 51624) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 19, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: August 31, 1994.

Brief description of amendments: These amendments relocate the requirements of TS 3/4.8.4.1, ''Primary Containment Penetration Conductor Overcurrent Protective Devices,'' to the Updated Final Safety Analysis Report and plant procedures.

Date of issuance: June 22, 1995.

Effective date: June 22, 1995.

Amendment Nos. 93 and 57.

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

Date of initial notice in **Federal Register:** November 9, 1994 (59 FR 55884) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 22, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: August 12, 1994, as supplemented by letter dated March 29, 1995.

Brief description of amendments: These amendments revise the action statements regarding emergency core cooling systems to allow continued operation in the event that the high pressure coolant injection system, one core spray subsystem and/or one low pressure coolant injection subsystem are inoperable.

Date of issuance: June 22, 1995.

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

Amendment Nos. 94 and 58.

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

Date of initial notice in **Federal Register:** October 12, 1994 (59 FR 51623). The March 29, 1995, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 22, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: August 31, 1994.

Brief description of amendments: The amendments permit the operability of one Low Pressure Coolant Injection subsystem of Residual Heat Removal while the subsystem is aligned and operating in the Shutdown Cooling Mode during Operational Conditions (OPCONs) 4 and 5.

Date of issuance: June 22, 1995.

Effective date: June 22, 1995.

Amendment Nos. 95 and 59.

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

Date of initial notice in **Federal Register:** November 9, 1994 (59 FR 55884). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 22, 1995.

No significant hazards consideration comments received: No.

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

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 application for amendments: November 18, 1994.

Brief description of amendments: The amendments revise the Reactivity Control System Technical Specification Limiting Conditions for Operation for boration flow paths and charging pumps by reducing the number of operable charging pumps required for boron addition in Mode 4 from two to one.

Date of issuance: June 12, 1995.

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

Amendment Nos. 169 and 151.

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

Date of initial notice in **Federal Register:** January 4, 1995 (60 FR 505). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 12, 1995.

No significant hazards consideration comments received: No.

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

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 application for amendments: June 29, 1994, as supplemented August 8, 1994, and May 2, 1995.

Brief description of amendments: The amendments increase the Technical Specification minimum volume of emergency diesel generator fuel oil contained in the Diesel Fuel Oil Storage Tanks at both units of the Salem station.

Date of issuance: June 20, 1995.

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

Amendment Nos. 170 and 152.

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

Date of initial notice in **Federal Register:** August 17, 1994 (59 FR 42346). The August 8, 1994, and May 2, 1995, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 20, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 6, 1995, as supplemented on May 5, 1995 and June 6, 1995.

Brief description of amendment: The amendment deletes a license condition that required the licensee to maintain a seismic monitoring network around the Monticello Reservoir.

Date of issuance: June 13, 1995.

Effective date: June 13, 1995.

Amendment No.: 124.

Facility Operating License No. NPF–12. Amendment revises the operating license.

Date of initial notice in **Federal Register:** March 29, 1995 (60 FR 16201). The May 5, 1995 and June 6, 1995 submittals provided supplemental information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 13, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 15, 1994; superseded March 7, 1995 (TS 350).

Brief Description of amendment: The amendments remove the frequencies specified in the Technical Specifications for performing audits and delete the requirement to perform the Radiological Emergency Plan, Physical Security Plan, and Safeguard Contingency Plan reviews.

Date of issuance: June 19, 1995.

Effective Date: June 19, 1995.

Amendment Nos.: 221, 236 and 195.

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

Date of initial notice in **Federal Register:** December 21, 1994 (59 FR 65823); superseded March 29, 1995 (60 FR 16202). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 19, 1995.

No significant hazards consideration comments received: None.

Local Public Document Room Location: Athens Public Library, South Street, Athens, Alabama 35611.

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

Date of application for amendments: April 6, 1995 (TS 95–02).

Brief description of amendments: The amendments add a limiting condition for operation that allows equipment to be returned to service under administrative control to perform operability testing and establishes the time interval to place an inoperable channel in the bypass condition.

Date of issuance: June 13, 1995.

Effective date: June 13, 1995.

Amendment Nos.: 202 and 192.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the technical specifications.

Date of initial notice in **Federal Register:** April 26, 1995 (60 FR 20530). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 13, 1995.

No significant hazards consideration comments received: None.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

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: April 6, 1995 (TS 95–05).

Brief description of amendments: The amendments revise the technical specifications by deleting Tables 3.6–1, 3.6–2, and 3.8–2 and referenced to them, incorporating related guidance and justification, and modifying the specification related to electrical equipment protective devices in accordance with Generic Letter 91–08.

Date of issuance: June 13, 1995.

Effective date: June 13, 1995.

Amendment Nos.: 203 and 193.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the technical specifications.

Date of initial notice in **Federal Register:** May 10, 1995 (60 FR 24919). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 13, 1995.

No significant hazards consideration comments received: None.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

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: April 6, 1995 (TS 95–06).

Brief description of amendments: The amendments remove the technical specification requirements related to crane travel over the spent fuel pool.

Date of issuance: June 14, 1995.

Effective date: June 14, 1995.

Amendment Nos.: 204 and 194.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the technical specifications.

Date of initial notice in **Federal Register:** April 26, 1995 (60 FR 20529). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 14, 1995.

No significant hazards consideration comments received: None.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

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

Date of application for amendment: October 28, 1994.

Brief description of amendment: The amendment removes the Neutron Monitoring System and Control Rod Position instrumentation from the Vermont Yankee Technical Specifications for post-accident monitoring and incorporates administrative changes.

Date of issuance: June 20, 1995.

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

Amendment No.: 145.

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

Date of initial notice in **Federal Register:** May 10, 1995 (60 FR 24922). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 20, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 9, 1994.

Brief description of amendments: These amendments modify the Chemical and Volume Control System and Safety Injection System Technical Specifications.

Date of issuance: May 31, 1995.

Effective date: May 31, 1995.

Amendment Nos. 199 and 199.

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

Date of initial notice in **Federal Register:** July 20, 1994 (59 FR 37089). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 31. 1995.

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 27th day of June 1995.

For the Nuclear Regulatory Commission.

Jack W. Roe,

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

[FR Doc. 95–16249 Filed 7–3–95; 8:45 am]

BILLING CODE 7590–01–P

Standard Technical Specifications (Revision 1): Availability

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) previously noticed the availability of five sets of improved Standard Technical Specifications (STS), Revision 0 that were issued on September 29, 1992 [57 FR 55602]. The NRC issued improved STS, Revision 0 for implementation by the volunteering leadplant licensees and placed copies in the NRC public document room. Subsequently, the NRC revised the improved STS (Revision 1) to incorporate additional comments from the Nuclear Steam Supply System (NSSS) owners groups and the NRC.

The STS for each NSSS vendor are as follows:

NUREG–1430, "Standard Technical Specifications, Babcock and Wilcox Plants"
NUREG–1431, "Standard Technical Specifications, Westinghouse Plants"
NUREG–1432, "Standard Technical Specifications, Combustion Engineering Plants"
NUREG–1433, "Standard Technical Specifications, General Electric Plants, BWR/4"
NUREG–1434, "Standard Technical Specifications, General Electric Plants, BWR/6"

The NRC staff operates the Tech Spec Plus Bulleting Board System (BBS) as a public service for anyone who wishes to obtain copies of electronic files of the STS. The NRC developed the STS with WordPerfect, version 5.1, software and has placed Revision 1 of the improved STS on the BBS in compressed form using "ZIP" data compression software to reduce the time required to download the files. The NRC BBS may be reached by telephone at 1–800–679–5784. Access to the BBS is available using a personal computer and modem with any standard communication software package. The BBS operates 24 hours a day at up to 9600 baud with communication parameters set at 8 data bits, no parity, and 1 stop bit (8–N–1). The system operator is Tom Dunning. He can be reached by telephone (voice) at (301) 415–1189, if assistance is needed.

Copies of the STS, Revision 1, are available for inspection or copying for a fee in the NRC Public Document Room, 2120 L Street NW., Lower Level of the Gelman Building, Washington, DC 20555. Requests for copies may be made by writing to the NRC Public Document Room or by facsimile at (202)–634–3343, or by telephone (202)–634–3273. Those requesting copies should identify the STS by NUREG number and title as noted above.

In addition, NUREG copies are available from the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013–7082.

FOR FURTHER INFORMATION CONTACT: Mary Lynn Reardon, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Telephone (301) 415–1177.

Dated at Rockville, Maryland, this 27th day of June, 1995.

For the Nuclear Regulatory Commission.

Christopher I. Grimes,

Chief, Technical Specifications Branch, Division of Project Support, Office of Nuclear Reactor Regulation.

[FR Doc. 95–16370 Filed 7–3–95; 8:45 am]

BILLING CODE 7590–01–M

RESOLUTION TRUST CORPORATION

Coastal Barrier Improvement Act; Property Availability; Millwood Estates, Clarke County, VA; Pine Island, Lee County, FL

AGENCY: Resolution Trust Corporation.

ACTION: Notice.

SUMMARY: Notice is hereby given that the properties known as Millwood Estates, located in Boyce, Clarke County, Virginia, and Pine Island, located in Pine Island, Lee County, Florida, are affected by Section 10 of the Coastal Barrier Improvement Act of 1990 as specified below.

DATES: Written notice of serious interest to purchase or effect other transfer of all or any portion of these properties may be mailed or faxed to the RTC until October 3, 1995.

ADDRESSES: Copies of detailed descriptions of these properties, including maps, can be obtained from or are available for inspection by contacting the following person: Mr. Dan Hummer, Resolution Trust Corporation, Atlanta Field Office, 245 Peacetree Center Avenue, NE., Marquis One Tower, 10th Floor, Atlanta, GA 30303, (404) 230–6594; Fax (404) 230–8159.

SUPPLEMENTARY INFORMATION: The Millwood Estates property is located at