to discuss matters the release of which would constitute a clearly unwarranted invasion of personal privacy per 5 U.S.C. 552b(c)(6).

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Mr. Sam Duraiswamy, Chief, Nuclear Reactors Branch (telephone 301/415–7364), between 7:30 A.M. and 4:15 P.M. EDT.

ACRS meeting notices, meeting transcripts, and letter reports are now available on FedWorld from the ''NRC MAIN MENU.'' Direct Dial Access number to FedWorld is (800) 303–9672; the local direct dial number is 703–321–3339.

Dated: June 27, 1996.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 96–16962 Filed 7–2–96; 8:45 am]

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UNITED STATES NUCLEAR REGULATORY COMMISSION

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from June 8, 1996, through June 21, 1996. The last biweekly notice was published on June 19, 1996 (61 FR 31171).

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

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

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

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

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

By August 2, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

Date of amendment request: May 1, 1996

Description of amendment request: The proposed amendment would modify Table 3.1.1, ''Reactor Protection System (SCRAM) Instrumentation Requirement,'' Table 3.2.C.1, ''Instrumentation that Initiates Rod Blocks,'' and Technical Specification 3/4.4, ''Standby Liquid Control.''

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

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

Note 7 to Table 3.1.1 and Note 6 to Table 3.2.C.1

The changes to Note 7 to Table 3.1.1 and the addition of Note 6 to Table 3.2.C.1 are proposed to clarify their requirements, the appropriate action to take, and their relationship to plant modes. This revised scram and rod block applicability is acceptable because control rods withdrawn from a core cell containing no fuel assemblies have a negligible impact on the reactivity of the core, and, therefore, these features are not required to be operable (i.e. provide the capability to scram). Provided all rods otherwise remain inserted, the RPS [Reactor Protection System] functions serve no purpose and are not required. In this condition, the required shutdown margin (Specification 3.3.A.1) and the required one-rod-out interlock (Specification 3.10.A) ensure that no event requiring the RPS or Rod Block will occur.

The Actions of Table 3.1.1 for inoperable equipment were previously revised in Amendment ι147 to be consistent with the improved STS [Standard Technical Specifications]. Action (A) requires fully inserting all insertable control rods in core cells containing one or more fuel assemblies. Since Specification 3.10.A requires all control rods to be fully inserted during fuel movement, the proposed applicable conditions cannot be entered while moving fuel. In addition, Specification 3.10.D used for controlling multiple control rod removal, requires all control rods in a 3X3 array centered on the CRDs [Control Rod Drive] being removed to be fully inserted and electrically disarmed and all other control rods fully inserted. The only possible action is control rod withdrawal, which is addressed by Action A.

Hence operating Pilgrim in accordance with the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Section 3/4.4

The proposed change involves reformatting, renumbering, and rewording of the existing Technical Specifications and Bases along with other changes to the Technical Specifications discussed above. The reformatting, renumbering, and rewording along with the other changes listed involves no technical changes to existing Technical Specifications, and does not impact initiators of analyzed events. It also does not impact the assumed mitigation of accidents or transient events. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change relocates requirements to other sections of the Technical Specifications, to plant procedures, or to the Technical Specifications BASES. The procedure change and BASES change processes require any changes that reflect plant design as described in the FSAR [Final Safety Analysis Report] be evaluated in accordance with 10 CFR 50.59. Since any changes will be evaluated per 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change provides more stringent requirements than previously existed in the Technical Specifications. The more stringent requirements will not result in operation that will increase the probability of initiating an analyzed event. If anything the new requirements may decrease the probability or consequences of an analyzed event by incorporating the more restrictive changes discussed above. The change will not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements will not alter the operation of process variables, structures, systems, or components as described in the safety analyses. Therefore, the change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change deletes the requirements for Standby Liquid Control (SLC) System operability during Hot Shutdown, Cold Shutdown, and Refueling. The SLC System is not assumed in the initiation of any previously evaluated events and therefore the proposed change will not increase the probability or consequence of a previously analyzed accident. The SLC System is not assumed to operate in the mitigation of any previously analyzed accidents which are assumed to occur during Hot Shutdown, Cold Shutdown or Refueling. This change will not result in operation that will increase the probability of initiating an analyzed event. This change will not alter assumptions relative to mitigation of an accident or alter the operation of process variables, structures, systems, or components as described in the safety analyses. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change adds an action for both SLC subsystems inoperable that delays the requirement to initiate plant shutdown immediately and allows time to recover at least one subsystem before subjecting the plant to a potentially unnecessary transient. Allowing a short period of time to recover one subsystem is acceptable because of the large number of independent control rods available to shut down the reactor and the diversity of means available to cause control rod insertion. This change will not alter assumptions relative to mitigation of an accident or alter the operation of process variables, structures, systems, or components as described in the safety analyses. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change deletes requirements for demonstrating operability of the redundant subsystems which eliminates excessive and unnecessary testing of safety significant equipment. This is consistent with guidance 10.1 of Generic Letter 93-05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirement for Testing During Power Operations''. The change does not affect the ability of the SLC system to perform on demand, and by actually lowering the number of demands to demonstrate operability, reduces the probability of equipment failure. Since the change will not alter assumptions relative to mitigation of an accident or alter the operation of process variables, structures, systems, or components as described in the safety analyses, the change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change replaces the requirement to verify B-10 enrichment concentration by test anytime boron is added to the solution and each refueling outage with verifying the enrichment prior to addition. Since enrichment of the solution in the tank cannot change by any other means but chemical addition, ensuring that only properly enriched material is available for addition is adequate to maintain enrichment at the required level. This change will not alter assumptions relative to mitigation of an accident or alter the operation of process variables, structures, systems, or components as described in the safety analyses. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Pilgrim Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Note 7 to Table 3.1.1 and Note 6 to Table 3.2.C.1

The changes to Note 7 to Table 3.1.1, and the addition of Note 6 to Table 3.2.C.1 are proposed to clarify their requirements, the appropriate action to take, and their relationship to plant modes. This revised scram and rod block applicability is acceptable because control rods withdrawn from a core cell containing no fuel assemblies have a negligible impact on the reactivity of the core, and, therefore, are not required to be operable. Provided all rods otherwise remain inserted, the RPS functions serve no purpose and are not required. In this condition, the required shutdown margin (Specification 3.3.A.1) and the required one-rod-out interlock (Specification 3.10.A) ensure that no event requiring the RPS or Rod Block will occur.

The Actions of Table 3.1.1 for inoperable equipment were previously revised in Amendment ı147 to be consistent with the improved STS. Action (A) requires fully inserting all insertable control rods in core cells containing one or more fuel assemblies. Since Specification 3.1O.A requires all control rods to be fully inserted during fuel movement, the proposed applicable conditions cannot be entered while moving fuel. In addition, Specification 3.10.D, used for controlling multiple control rod removal, requires all control rods in a 3X3 array centered on the CRDs being removed to be fully inserted and electrically disarmed and all other control rods fully inserted. The only possible action is control rod withdrawal, which is addressed by Action A. Hence, operating Pilgrim in accordance with the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Section 3/4.4

The proposed change involves reformatting, renumbering, and rewording of the existing Technical Specifications and Bases along with other changes to the Technical Specifications discussed above. The reformatting, renumbering, and rewording along with the other changes listed involves no technical changes to existing Technical Specifications. These changes are administrative and do not impact the assumed mitigation of accidents or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change relocates requirements to other Technical Specification sections, to plant procedures, or to the Technical Specification BASES. Relocating requirements will not alter the plant configuration (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. Relocating requirements will not impose different requirements and adequate control of information will be maintained. Relocating requirements will not alter assumptions made in the safety analysis and licensing basis. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes make some existing requirements more restrictive and add additional requirements to the Technical Specifications but will not alter the plant configuration (no new or different type of equipment will be installed) or change methods governing normal plant operation. These changes do impose different requirements, however, they are consistent with assumptions made in the safety analyses. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change relaxes the modes of applicability for the SLC. Relaxing the applicability will not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Pilgrim Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

Note 7 to Table 3.1.1 and Note 6 to Table 3.2.C.1

This revised scram and rod block applicability is acceptable because control rods withdrawn from a core cell containing no fuel assemblies have a negligible impact on the reactivity of the core, and, therefore, are not required to be operable (provide a scram). Provided all rods otherwise remain inserted, the RPS functions serve no purpose and are not required. In this condition, the required shutdown margin (Specification 3.3.A.1) and the required one-rod-out interlock (Specification 3.10.A) ensure that no event requiring the RPS or Rod Block will occur.

The Actions of Table 3.1.1 for inoperable equipment were previously revised in

Amendment ı147 to be consistent with the improved STS. Action (A) requires fully inserting all insertable control rods in core cells containing one or more fuel assemblies. Since Specification 3.10.A requires all control rods to be fully inserted during fuel movement, the proposed applicable conditions cannot be entered while moving fuel. In addition, Specification 3.10.D, used for controlling multiple control rod removal, requires all control rods in a 3X3 array centered on the CRDs being removed to be fully inserted and electrically disarmed and all other control rods fully inserted. The only possible action is control rod withdrawal, which is adequately addressed by Action A.

Therefore, operating Pilgrim in accordance with the proposed changes will not involve a significant reduction in a margin of safety.

Section 3/4.4

The administrative changes involve no technical changes. These proposed changes will not reduce a margin of safety because there is no impact on any safety analysis assumptions. Also, because the change is administrative in nature, no question of safety is involved. Therefore, these changes do not involve a significant reduction in a margin of safety. The change relocates requirements to other Technical Specification sections, to plant procedures, or to the Technical Specification BASES. These changes will not reduce a margin of safety since there is no impact on any safety analysis assumptions. In addition, the requirements to be transposed are the same as the existing Technical Specifications. Since any changes to plant procedures and Technical Specification BASES are required to be evaluated per 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed. Therefore, these changes will not involve a significant reduction in a margin of safety.

The addition of new requirements and making existing ones more restrictive either increases or does not affect the margin of safety. These changes do not impact any safety analysis assumptions. As such, no question of safety is involved. Therefore, these changes will not involve a significant reduction in a margin of safety.

The proposed change would remove a backup (in the Hot Shutdown, Cold Shutdown, and Refueling Modes) to the available systems for reactivity control; however, this backup is not considered in the margin of safety when determining the required reactivity for shutdown and refueling events. This change will have no impact on any safety analysis assumptions. As such, no question of safety is involved. Therefore, this change does not involve a significant reduction in a margin of safety.

The SLC system is not assumed to function in any DBA or transient and is not the primary success path of a safety sequence analysis. It is a backup to the CRD scram function, therefore, allowing a short period of time to recover one subsystem will have no impact on any safety analysis assumptions. As such, no question of safety is involved. Therefore, this change does not involve a significant reduction in a margin of safety.

The change does not alter the requirements for enrichment/ concentration of the boron solution necessary to satisfy 10 CFR 50.62. Since enrichment of the solution in the tank cannot change by any other means but chemical addition, this change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Jocelyn A. Mitchell, Acting Director

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

Date of amendments request: November 15, 1995

Description of amendments request: The proposed amendments would revise the Technical Specifications (TS) to alter the wording of TS 4.8.2.5.a in accordance with the guidance of Generic Letter (GL) 91-09, ''Modification of Surveillance Interval For The Electrical Protection Assemblies In Power Supplies For The Reactor Protection System.''

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

1. The proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed change does not alter the design, function, or operation of the EPAs [Electrical Protective Assemblies]. The proposed amendments modify the surveillance requirement for an electrical protective device on the Reactor Protection System [RPS]. The RPS-EPA units are designed to protect RPS equipment from abnormal operating voltage or frequency. The proposed change will preclude the need to test the RPS-EPA units during power operation. This will eliminate the potential for reactor scrams and Group isolations during performance of the surveillance, thus, preventing unwarranted challenges to safety systems. The proposed change does not affect any accident precursor or initiator. Therefore, the probability of an accident is not affected by the proposed change. The proposed amendments do not affect the operability of the RPS-EPA units. The proposed change does not affect the ability of the Reactor Protection System to maintain the integrity of the fuel cladding, protect the reactor coolant pressure boundary, or limit the amount of energy released to primary containment. Therefore, the consequences of an accident is not affected by the proposed change.

2. The proposed amendments do not create the possibility of a new or different kind of accident from any accident previously evaluated. As stated above, these proposed amendments do not alter the design, functions, or operation of the EPAs. The RPS relay trip logic remains protected from power supplies operating with abnormal voltage or frequency. Additionally, the redundancy of this protection is not changed.

Thus, the proposed amendments do not create the possibility of a new or different kind of accident.

3. The proposed amendments do not involve a significant reduction in a margin of safety because the benefit to safety by reducing the frequency of testing during power operation and attendant possible challenges to safety systems more than offsets any risk to safety from relaxing the surveillance requirement to test the EPAs during power operation. The testing of each EPA channel involves a dead-bus transfer and the momentary interruption of power results in a half scram and half isolation. Generic Letter 91-09 notes that many plants have encountered problems with the reset of the half trip resulting in inadvertent scrams and group isolations that challenge safety systems during power operation. Eliminating EPA testing at power operation increases the margin of safety by eliminating the potential for trips due to testing that challenge safety systems. An insignificant reduction in the margin of safety is introduced by increasing the test interval up to a maximum of a refuel cycle which will produce a small increase in risk that an inoperable EPA would not be detected. The elimination of potential challenges to safety systems provides a safety benefit that offsets the increased risks of component failure.

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

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

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

NRC Project Director: Eugene V. Imbro

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 6, 1996

Description of amendment request: The proposed change would revise technical specifications (TS) Section 4.2.3 to allow the licensee to defer the ultrasonic inspection of the reactor coolant pump flywheel for one operating cycle.

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 safety function of the Reactor Coolant Pump (RCP) flywheel is to provide a coastdown period during which the RCPs would continue to provide reactor coolant flow to the core after a loss of power to the RCPs. The maximum loading on the RCP motor flywheel results from overspeed following a large break Loss of Coolant Accident (LOCA). The estimated maximum obtainable speed in the event of a Reactor Coolant System (RCS) piping break was established conservatively, and the proposed one-time change does not affect that analysis.

The RCP flywheels have been carefully designed and manufactured from high quality steel. Twenty-two inspections have been performed at HBRSEP, Unit No. 2 over the past 25 years and no indications have been discovered that would affect the integrity of the flywheel. The Westinghouse Owners Group (WOG) has performed an extensive study documented in WCAP-14535, ''Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination,'' that includes an evaluation of industry experience, a stress and fracture evaluation, and a risk assessment, and has concluded that RCP flywheel inspections may be safely eliminated.

Reduced coastdown times due to a single failed flywheel would not place the plant in an unanalyzed condition since a locked rotor (i.e., an instantaneous coastdown) is analyzed in the Updated Final Safety Analysis Report (UFSAR). The proposed change also does not increase the amount of radioactive material available for release or modify any systems used for mitigation of releases during an accident. Therefore, the proposed change does not involve an increase in the probability of 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 will not change the design, configuration, or method of operation of the plant. Therefore, the proposed change will not create the possibility of a new kind of accident from any previously evaluated.

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

The RCP flywheels have been carefully designed and manufactured from high quality steel. Twenty-two inspections have been performed at HBRSEP, Unit No. 2 over the past 25 years and no indications have been discovered that would affect the integrity of the flywheel. The Westinghouse Owners Group (WOG) has performed an extensive study documented in WCAP-14535, ''Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination,'' that includes an evaluation of industry experience, a stress and fracture evaluation, and a risk assessment, and has concluded that RCP flywheel inspections may be safety eliminated. The proposed change would only result in a one-time deferral of the scheduled inspection for one operating cycle. In consideration of the historical integrity of the HBRSEP, Unit No. 2 RCP flywheels, the industry experience, the results of the WOG study, and the deferral of the risk of RCP flywheel damage during disassembly and inspection, we conclude that a one operating cycle deferral of the scheduled RCP flywheel inspection will not result in a reduction in the margin of safety.

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

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

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

NRC Project Director: Eugene V. Imbro

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

Date of amendment request: May 31, 1996

Description of amendment request: The proposed amendment would change the plant Technical Specifications (TS) Table 3.3-7, Seismic Monitoring Instrumentation, and TS Table 4.3-4, Seismic Monitoring Instrumentation Surveillance Requirements, to correct the location described for one of the three Triaxial Peak Accelerograph Recorders.

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

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

These recorders are passive components which serve only a recording function. They can neither initiate an accident nor serve to mitigate accident consequences. The proposed change serves only to correct the location, commensurate with design documents, for one of the three recorders described in the Technical Specifications. Accordingly, this change is administrative in nature. Therefore, there would be no increase in the probability or consequences of an accident previously evaluated.

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

The proposed correction is an administrative change to correct the location of a recorder currently described in the Technical Specifications. No physical alterations to plant equipment are being made, and there will be no changes that alter how any safety-related system performs its function. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Technical Specification Bases 3/4.3.3.3 specify the acceptance level for seismic instrumentation as ''consistency'' with the recommendations of Regulatory Guide 1.12. Since the regulatory guide states only that one recorder should be provided at a ''selected location on the reactor piping,'' it is not material whether it is installed on Loop 1 versus Loop 2. Therefore, the proposed change does not affect a margin of safety as defined in the Bases to 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: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605

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

NRC Project Director: Eugene V. Imbro

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

Dates of amendment request: December 18, 1995, May 3 and June 11, 1996

Description of amendment request: The licensee proposed to change the Turkey Point Units 3 and 4 Technical Specifications (TS) to uprate the core thermal output of Turkey Point Units 3 and 4 from 2200 MWt to 2300 MWt. The proposed TS changes were divided into eight groups. The submittal included a ''No Significant Hazards'' evaluation for each of the eight groups. The groupings are as follows:

TS changes associated with the uprated power level, the revised core safety limits, revised DNB [departure from nucleate boiling] parameters, Engineered Safety Features Actuation System (ESFAS) and reactor trip setpoint changes, and Reactor Coolant Pump (RCP) Breaker Position Trip, were evaluated together. The safety of these proposed changes were verified by the accident analyses that were completed in support of the uprated power.

TS changes associated with reducing the SI [safety injection] pump discharge head requirement and increasing usable volume requirements for the Demineralized Water Storage Tank (DWST) and the Condensate Storage Tank (CST) were addressed together.

TS changes associated with pressurizer and main steam safety valve (MSSV) setpoint tolerance increases were assessed together.

TS changes associated with operation at reduced power with inoperable MSSVs were assessed separately.

TS changes associated with the service period for heatup and cooldown pressure-temperature limit curves were assessed together.

The Surveillance Requirement change for the emergency containment cooling [ECC] unit operability was handled separately since this was a design change that required extensive evaluations.

TS change associated with the methyl iodide removal efficiency in the Control Room Emergency Ventilation System was assessed separately.

All LOCA [loss-of-coolant accident] related changes dealing with the peaking factor increase, COLR [core operating limit report] changes, Evaluation Model references, and relocation of peaking factors from the TS and subsequent inclusion in the COLR were included in one ''No Significant Hazards'' evaluation. All of the items are closely related since the LOCA analysis is performed to ensure peaking factor acceptability.

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.

LICENSE CONDITION, RATED THERMAL POWER, CORE SAFETY LIMITS, REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS, ESFAS INSTRUMENTATION TRIP SETPOINTS, DNB PARAMETERS AND RCP BREAKER POSITION TRIP

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

The proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated because operation with these revised values will not cause any design or analysis acceptance criteria to be exceeded. The structural and functional integrity of all plant systems are unaffected. The overtemperature Delta T and overpower Delta T reactor trip functions as well as ESFAS functions are part of the accident mitigation response and are not accident initiators. All proposed changes have been assessed and no design and analysis acceptance criteria have been exceeded. Therefore the probability of occurrence previously evaluated is not affected.

The proposed changes do not affect the integrity of the fission product barriers utilized for mitigation of dose consequences as a result of an accident. Dose consequences were reviewed and reanalyzed (as needed) and found acceptable. Therefore, the probability or consequences of an accident previously evaluated are not significantly increased.

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

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because their effects do not affect accident initiation sequences. All new operating configurations have been evaluated and no new limiting single failures have been identified. In addition, no new failure modes have been identified. Therefore, it is concluded that no new or different kind of accident from any accident previously evaluated has been created as a result of these revisions.

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

The proposed changes do not involve a reduction in a margin of safety because the margin of safety associated with these parameters as verified by the results of the accident analyses, are within acceptable limits. All transients impacted have been analyzed and have met the applicable accident analyses acceptance criteria (e.g., DNBR [departure from nucleate boiling ratio], RCS [reactor coolant system] pressure, secondary side pressure, etc.). The margin of safety required for each affected safety analysis is maintained. The adequacy of the revised Technical Specifications values has been confirmed such that there is no reduction in the margin of safety. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

AVAILABLE VOLUME CHANGE FOR CONDENSATE STORAGE TANK (CST) AND DEMINERALIZED WATER STORAGE TANK (DWST), AND REDUCED SAFETY INJECTION (SI) PUMP DISCHARGE HEAD REQUIREMENT.

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

The revised tank volumes and SI head requirements have been evaluated with respect to system performance and analysis impacts. All accident analysis acceptance criteria continue to be met. The design function of all affected systems have been reviewed and all system design criteria continue to be met. The structural and functional integrity of the affected systems are unaffected. These changes are not initiators for any accident and therefore the probability of occurrence of an accident previously evaluated has not increased.

The proposed changes do not affect the integrity of the fission product barriers for mitigation of dose consequences. All dose consequences remain well within the 10 CFR 100 limits. Therefore there is no increase in the probability or consequences of an accident previously evaluated.

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

The revised tank volumes and SI head requirements do not create the possibility of a new or different kind of accident from any accident previously evaluated because these modifications do not affect accident initiation sequences. No new operating configuration is being imposed by the adjustments that would create a new failure scenario. In addition, no new failure modes or limiting single failures have been identified. Therefore, it is concluded that no new or different kind of accident from any accident previously evaluated have been created as a result of these revisions.

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

The proposed changes do not involve a reduction in a margin of safety because the margin of safety associated with these parameters, as verified by the results of the accident analyses and system evaluations, are within acceptance limits. The margin of safety required for each affected safety analysis is maintained. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

PRESSURIZER AND MAIN STEAM SAFETY VALVE SETPOINT TOLERANCES

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

The revised tolerances for main steam safety valves and pressurizer safety valves do not involve an increase in the probability or consequences of an accident previously evaluated because operation with these revised values will not cause any design or analytical acceptance criteria, such as those applicable to primary and secondary side pressures to be exceeded. The structural and functional integrity of the valves are unaffected by this proposed change. The tolerance changes do not initiate or cause initiation of any transient. Therefore, the probability of occurrence previously evaluated is not affected.

The changes do not affect the integrity of the fission product barriers utilized for dose consequence mitigation. Therefore, the probability or consequences of an accident previously evaluated is not increased.

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

The revised valve tolerances do not create the possibility of a new or different kind of accident from any accident previously evaluated because the tolerances do not affect accident initiation sequences. No new operating configuration is being imposed by the tolerances that would create a new failure scenario. In addition, no new failure modes or limiting single failures have been identified. Therefore, it is concluded that no new or different kind of accident from any accident previously evaluated have been created as a result of these revisions.

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

The changes to valve tolerances do not involve a reduction in a margin of safety because the margin of safety associated with the MSSVs and the pressurizer safety valves, as verified by the results of the accident analyses and valve evaluations, are within acceptable limits. Transients impacted by this change have been analyzed and have met the applicable accident analyses acceptance criteria, such as those applicable to primary and secondary side pressure. The margin of safety required for each affected safety analysis is maintained. This conclusion is not changed by the valve tolerances for the main steam safety valves and the pressurizer safety valves. Therefore, the changes do not involve a significant reduction in the margin of safety.

OPERATION AT REDUCED POWER WITH INOPERABLE MAIN STEAM SAFETY VALVES

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

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed maximum allowable power level values will ensure that the secondary side steam pressure will not exceed 110 percent of the design pressure following a Loss of Load/Turbine Trip event, when one or more main steam safety valves (MSSVs) are declared inoperable. The proposed change will not impact the classification of the Loss of Load/Turbine Trip event as a Condition II probability event (faults of moderate frequency) per ANSI - N18.2, 1973. Accordingly, since the proposed maximum allowable power level will maintain the capability of the MSSVs to perform their pressure relief function associated with a Loss of Load/Turbine Trip event, there will be no effect on the probability or consequences of an accident previously evaluated.

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

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve any change to the configuration of any plant equipment, and no new failure modes have been defined for any plant system or component. The proposed maximum allowable power level as specified in TS Table 3.7-1 will improve the capability of the MSSVs to perform their pressure relief function to ensure the secondary side steam pressure does not exceed 110 percent of design pressure following a Loss of Load/ Turbine Trip event. Therefore, since the function of the MSSVs is improved by the proposed changes, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

The proposed changes to the Technical Specifications do not involve a significant reduction in a margin of safety. The algorithm methodology used to calculate the maximum allowable power level is conservative and bounding since it is based on a number of inoperable MSSVs per loop; i.e., if only one MSSV in one loop is out of service, the required action to reduce power to the maximum allowable power level would be the same as if one MSSV in each loop were out of service. Another conservatism with the algorithm methodology is with the assumed minimum total steam flow rate capability of the operable MSSVs. The assumption is that if one or more MSSVs are inoperable per loop, the inoperable MSSVs are the largest capacity MSSVs, regardless of which capacity MSSVs are actually inoperable.

Therefore, since the maximum allowable power level calculated for the proposed changes using the algorithm methodology are more conservative and ensure that 110 percent of secondary side steam pressure is not exceeded following a Loss of Load/ Turbine Trip event, this proposed license amendment will not involve a significant reduction in a margin of safety.

SERVICE PERIOD FOR HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

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

Calculation of the service period for the heatup and cooldown curves does not involve an increase in the probability or consequences of an accident previously evaluated because the calculations were completed to verify the adequacy of the existing curves and to determine an appropriate service period. The use of approved methods and the acceptable results have shown that no design or analysis criteria are changed. The structural and functional integrity of the reactor vessel has been verified.

No fission product barriers or inputs to dose analyses are adversely affected by these calculations and reverification of the existing heatup/cooldown curves. Therefore, the probability or consequences of an accident previously evaluated are not increased.

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

The revised service period does not create the possibility of a new or different kind of accident from any accident previously evaluated because the recalculation of an acceptable service period does not affect accident initiation sequences. No new operating configuration is being imposed by the calculations that would create a new failure scenario. In addition, no new failure modes or limiting single failures have been identified. Therefore, the types of accidents defined in the UFSAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation. Therefore, it is concluded that no new or different kind of accident from any accident previously evaluated have been created as a result of these revisions.

(3) Operation of the facility in accordance with the proposed license amendments would not involve a significant reduction in a margin of safety.

Calculations were performed to determine the service period appropriate for the existing curves. The changes to service period do not involve a reduction in a margin of safety because the margin of safety associated with the heatup/cooldown curves, as verified by the results of the analyses, are unchanged. Therefore, the proposed change to the service period does not involve a significant reduction in the margin of safety.

MODIFICATION TO SURVEILLANCE REQUIREMENT FOR EMERGENCY CONTAINMENT COOLING SYSTEM

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

The purpose of the ECC units is to help mitigate the consequences of an accident (i.e., to help maintain the containment pressure and temperature within their design values following a design basis accident). The ECC units do not operate during normal operation of the plant. Failure of the ECC units would not initiate a plant transient or accident. Therefore, the proposed change involving the ECC units would not affect the probability of occurrence of an accident previously evaluated.

Evaluations demonstrate that, with two ECC units operating during a LOCA or MSLB [main steamline break], the containment pressure and temperature will be maintained within their design values. These evaluations also demonstrate that, with two ECC units operating during a LOCA or MSLB, the temperature of the CCWS [component cooling water system] will be maintained within its design temperature. Therefore, the proposed change involving the ECC units would not affect the consequences of an accident previously evaluated.

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

The purpose of the ECC units is to mitigate design basis accidents, and failure of the ECC units would not cause a plant transient or accident. Furthermore, a single failure of an ECC unit during a LOCA or MSLB would not lead to a new or different kind of accident. Although the revised Technical Specifications require two ECC units to start automatically on a LOCA signal, they would also require that all three ECC units be operable. On a single failure of an operating ECC unit, there would be sufficient time to start the standby ECC unit to accomplish the design function of the ECC system. Therefore, the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change in the actuation logic of the ECC units would not cause either the containment pressure and temperature or the CCWS temperature to exceed their design values. While the energy released into containment and subsequently transferred to the CCWS will increase as a result of the thermal uprate, this increase is insignificant and will not result in either the containment or CCWS exceeding a design limit. Therefore, the proposed change would not affect the margin of safety.

CONTROL ROOM EMERGENCY VENTILATION SYSTEM

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

The proposed change does not affect the integrity of the fission product barriers utilized for mitigation of dose consequences as a result of an accident. Only the iodide removal efficiency of the control room emergency ventilation system is increased, and this change is in the conservative direction.

To assure consistency between testing efficiency and analysis assumptions for post-accident control room doses, the methyl iodide removal efficiency required to be demonstrated by laboratory test, is being increased from 90% to 99%. This increase in testing efficiency is consistent with the recommendations set by the NRC staff in Regulatory Guide 1.52 to support analysis efficiencies for elemental iodine and methyl iodide removal of 95%, respectively. Testing performed to verify methyl iodide removal efficiency will be performed under conditions representative of the control room environment.

Since this change in removal efficiency is in the conservative direction, plant safety will not be adversely impacted.

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

The proposed change to the control room emergency ventilation system iodide removal efficiency does not create the possibility of a new or different kind of accident from any accident previously evaluated because operation of the control room emergency ventilation system is not identified in any accident initiation sequence. The system is provided to minimize operator exposure to airborne radioactivity released as a result of an accident. The new operating configuration has been evaluated and no new limiting single failures have been identified as a result of the proposed modification. Therefore, it is concluded that no new or different kind of accidents from any accident previously evaluated have been created as a result of these revisions.

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

The proposed changes do not involve a reduction in the margin of safety because the margin of safety associated with this change is in the conservative direction. Thus, plant safety will not be adversely impacted and the margin of safety required for the affected safety analysis is maintained. The adequacy of the revised Technical Specification values to maintain the plant in a safe operating condition has been confirmed, since the testing will be done to a more conservative criteria (i.e., 99% efficiency). Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

RELOCATION OF $F_Q(Z)$ [HEAT FLUX HOT CHANNEL FACTOR] AND F Delta H [NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR] LIMITS FROM TECHNICAL SPECIFICATIONS TO CORE OPERATING LIMITS REPORT AND EDITORIAL CORRECTIONS

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

The relocation of the values for F_Q and F Delta H from the Technical Specifications to the Core Operating Limits Report is administrative in nature and has no impact on the probability or consequences of any Design Bases Event (DBE) occurrence which was previously evaluated. The determination of the F_Q and F Delta H limits will be performed using methodology approved by the NRC and poses no significant increase in the probability or consequences of any accident previously evaluated.

The changes being proposed as editorial in nature do not affect assumptions contained in the safety analyses, the physical design and/or operation of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. Therefore, these proposed changes do not affect the probability or consequences of accidents previously analyzed.

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

The relocation of the F_Q and F Delta H limits from the Technical Specifications to the Core Operating Limits Report is administrative in nature and has no impact, nor does it contribute in any way to the possibility of a new or different kind of accident from any accident previously evaluated.

The determination of the F_Q and F Delta H limits will be performed using NRC-approved methodology and are submitted to the NRC as a revision to the COLR to allow the NRC staff to trend peaking factors. The Technical Specifications will continue to require operation within the required core operating limits and appropriate actions will be taken if the F_Q and F Delta H limits are exceeded. Therefore, the proposed amendments does not in any way create the possibility of a new or different kind of accident from any accident previously evaluated.

The editorial changes proposed are administrative in nature and do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the facility, nor do they affect Technical Specifications that preserve safety analysis assumptions. Therefore, these changes do not create the possibility of a new or different kind of accident.

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

The relocation of the F_Q and F Delta H limits from the Technical Specifications to the Core Operating Limits Report is administrative in nature and has no impact on the margin of safety. The determination of the F_Q and F Delta H limits will be performed using methodology approved by the NRC and does not constitute a significant reduction in the margin of safety.

The supporting Technical Specification values are defined by the accident analyses which are performed to conservatively bound the operating conditions defined by the Technical Specifications. Performance of analysis and evaluation have confirmed that the operating envelope defined by the Technical Specifications continues to be bounded by the analytical basis, which in no case exceeds the acceptance limits. Therefore, the margin of safety provided in the analyses in accordance with the acceptance limits is maintained and not significantly reduced.

The changes being proposed as editorial in nature do not relate to or modify the safety margins defined in, and maintained by the Technical Specifications. Therefore, the proposed changes which correct administrative errors and clarify existing Technical Specification requirements do not involve any reduction in a margin of safety.

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

Local Public Document Room location: 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: Frederick J. Hebdon

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

Dates of amendment request: April 19, 1996, May 10, 1996, and May 28, 1996

Description of amendment request: The licensee proposed to change the Turkey Point Units 3 and 4 Technical Specifications (TS) to address frequency extension for actions required on a periodic basis, delete the separate notification requirement for an inoperable startup transformer, and allow the operating RHR loop to be removed from operation during refueling operations under certain conditions.

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

The proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed amendments are purely administrative in nature. These amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated because they do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. Therefore, the proposed changes do not affect the probability or consequences of accidents previously analyzed.

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

The use of the modified specifications can not create the possibility of a new or different kind of accident from any previously evaluated since the proposed amendments will not change the physical plant or the modes of plant operation defined in the facility operating license. No new failure mode is introduced due to the administrative changes and clarifications, since the proposed changes do not involve the addition or modification of equipment nor do they alter the design or operation of affected plant systems, structures, or components.

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

The operating limits and functional capabilities of the affected systems, structures, and components are unchanged by the proposed amendments. The modified specifications which correct administrative errors and clarify existing Technical Specification requirements do not significantly reduce any of the margins of safety.

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

Local Public Document Room location: 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: Frederick J. Hebdon

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, 1996

Description of amendment request: The proposed amendment would revise the technical specifications surveillance requirement (SR) 3.8.3.4 to specify a 5-start pressure for the air receivers associated with the Division III, High Pressure Core Spray emergency diesel generator.

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

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

The purpose of the proposed Technical Specification change is to establish consistency between the basis for the air start pressure required for the Division I and II diesels and the value required for the Division III diesel. The value of 160 psig currently specified in SR 3.8.3.4 is representative of a 5-start value for the Division I and II diesels, however, this value is not representative of a 5-start for the Division III diesel. While the 160 psig value does serve to satisfy the requirements of 10 CFR 50.36 with regard to maintaining the lowest functional level required for the Division III diesel to perform its design safety function, the current value does not serve to maintain the design margin utilized when sizing the air receivers for the purpose of satisfying the Standard Review Plan guidance contained in section 9.5.6 (NUREG-0800 Revision 2).

The proposed value fully complies with the guidance provided in NUREG-0800 and is more conservative than the value currently included in the Technical Specifications. The proposed value is well within the capability of the air system's design and will not subject the air system to excessive pressures or undue cycling of the system's compressors. The proposed change has no effect on the probability of an accident as diesel generators have no bearing on the initiation of any analyzed event. In addition, the capability of the Division III diesel to perform its design basis function (i.e., starting, accelerating to rated speed and voltage, and connecting to its respective bus within 13 seconds) is not affected by this change. The ability of the diesel to support the mitigation of analyzed accidents is not affected and hence the consequences of any analyzed event are not affected. Therefore, the proposed change does not increase the probability or the consequences of previously analyzed accidents.

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

The proposed change does not introduce any new failure modes. All of the affected components remain within their applicable design limits. In addition, the environmental qualification of any plant equipment is not adversely affected by the proposed change. Since the performance of this system is not adversely affected by this change and the design margins of this system are not challenged in a manner differently than previously analyzed, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change raises the required starting air pressure for the Division III above that currently required by the Technical Specifications to establish consistency between the basis of the Division III value with the value used for the Division I and II diesels. Issuance of the proposed change will establish a 5 start air receiver pressure for each of the three safety-related diesels at

River Bend. While the proposed value is slightly less than the 5 start value discussed in River Bend's SER, the proposed value is supported by the River Bend site-specific test data and does not adversely affect existing analyses or system performance. Therefore, the proposed change does not result in 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: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803

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

NRC Project Director: William D. Beckner

Nebraska Public Power District, Docket No. 50-298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: June 6, 1996, as supplemented by letters dated June 7 and 9, 1996

Description of amendment request: The proposed amendment would revise the technical specification Limited Safety System Setting for the MINIMUM CRITICAL POWER RATIO (MCPR) for dual recirculation loop operation and for single recirculation loop 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. Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

The purpose of the Safety Limit Minimum Critical Power Ratio (SLMCPR) is to provide statistical confidence that less than 0.1% of the fuel rods in a core would experience transition boiling during the most limiting analyzed Anticipated Operational Occurrence (transient). While transition boiling in a BWR does not in and of itself signal the onset of fuel cladding failure, this criterion has been selected as a conservative and convenient parameter for the evaluation of fuel designs. Therefore, while this safety limit does not provide any control over either the probability or consequences of any accident previously evaluated, it does ensure that evaluated transients remain within NRC-approved criteria. Revision of the SLMCPR will establish in the CNS Technical Specifications a valid limit, based on the NRC approved GESTAR II methodology using cycle-specific inputs. This change will result in the input of more restrictive core operating limits into the plant process computer, ensuring that CNS will be operated within the constraints of the new SLMCPR limits of 1.07 for dual recirculation loop operation, and 1.08 for single recirculation loop operation. No plant hardware modifications are associated with this change. Therefore, since this proposed change will not change the physical configuration of the plant, nor result in operational changes which invalidate assumptions used in any CNS accident analysis, this change does not involve an increase in the probability or consequences of any accident previously evaluated.

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

This change revises the SLMCPR values in the CNS Technical Specifications in accordance with a cycle specific analysis performed for the remainder of the current cycle. The SLMCPR ensures that less than 0.1% of the fuel rods in a core would experience transition boiling during the most limiting Anticipated Operational Occurrence. Increasing the SLMCPR from 1.06 to 1.07 for dual recirculation loop operation and from 1.07 to 1.08 for single recirculation loop operation will ensure that the specified statistical confidence will be met for all analyzed transients. This change does not involve any plant hardware changes. The only operational changes will be the institution of appropriate thermal restrictions on reactor core operation in accordance with the SLMCPR changes. Therefore, this proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change create a significant reduction in the margin of safety?

This change will establish in the CNS Technical Specifications, SLMCPR values that ensure the margin of safety to the NRC approved Anticipated Operational Occurrence evaluation acceptance criteria will be met. Increasing the SLMCPR institutes more restrictive thermal limitations on core operation. The change of the SLMCPR from 1.06 to 1.07 for dual recirculation loop operation, and from 1.07 to 1.08 for single loop operation will ensure that the acceptance criteria for evaluated transients will continue to be met, and that the appropriate limit is reflected in the CNS Technical Specifications. Therefore, this proposed change does not create a reduction in the margin of safety.

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

Local Public Document Room location: Auburn Memorial Library, 1810 Courthouse Avenue, Auburn, NE 68305

Attorney for licensee: Mr. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602-0499

NRC Project Director: William D. Beckner

Niagara Mohawk Power Corporation, Docket No. 50-410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of amendment request: May 15, 1996

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3/4.3.2, ''Isolation Actuation Instrumentation,'' to establish a range of allowable and trip setpoints for high temperature (varying as a function of ambient temperature) in the Main Steam Line Tunnel Lead Enclosure Area. Specifically, a new TS Figure 3.3.2-1 would be added to provide a curve of allowable temperature values and a curve of trip temperature setpoints, both plotted over a range of ambient temperatures. The new Figure would be referenced by Table 3.3.2-2 at item 1.d.3 (High Temperature Main Steam Line Tunnel Lead Enclosure Trip Function) by a new footnote stating:

The trip setpoint and allowable value for a channel may be established based on Figure 3.3.2-1, if:

a. The actual ambient temperature readings for all operable channels in the Lead Enclosure Area are equal to or greater than the ambient temperature used as the basis for the setpoint, and

b. The absence of steam leaks in the Main Steam Line Tunnel Lead Enclosure Area is verified by visual inspection prior to increasing a channel setpoint, and

c. A surveillance is implemented in accordance with Note (d) of Table 4.3.2.1-1.

Similarly, TS Surveillance Table 4.3.2.1-1 would be supplemented at item 1.d.3 (High Temperature Main Steam Line Tunnel Lead Enclosure) with a new footnote stating:

(d) In addition to the normal shift channel check, if a channel setpoint has been established using Figure 3.3.2-1, then once per shift, the actual ambient temperature reading for all operable channels in the Lead Enclosure Area shall be verified to be equal to or greater than the ambient temperature used as the basis for the setpoint.

Basis for proposed no significant hazards consideration determination: The main steam tunnel high temperature isolation actuation instrumentation is part of the Leak Detection System (LDS). It is used to detect leakage early at 25 gallons per minute (gpm) and initiate signals to automatically close the Main Steam Isolation Valves before a pipe break could occur. The existing temperature setpoints for the tunnel lead enclosure are based upon transient analyses for steam leaks in the steam tunnel utilizing winter temperatures as an initial condition. The licensee finds that a change is needed because actual temperatures in the tunnel, especially during the summer, are approaching the setpoints when steam leakage is not occurring. Under the present conditions, a minor disturbance in the turbine building ventilation system could cause an unwarranted isolation actuation at full power with resulting Main Steam Isolation Valve closure and reactor scram.

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 operation of NMP2 [Nine Mile Point Unit 2] in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The LDS instrumentation in the main steam line tunnel isolates the Main Steam Isolation Valves upon sensing a steam leak of 25 gpm. For an elevated ambient temperature in the Lead Enclosure area, a setpoint established using the proposed Figure 3.3.2-1 ensures that the Main Steam Isolation Valves continue to receive an isolation signal upon sensing a steam leak of 25 gpm. Verifying the absence of any steam leak in the area prior to raising any temperature instrument setpoint ensures that the ability to sense a 25 gpm leak is not compromised by an increased ambient temperature resulting from a smaller steam leak. The periodic surveillance to verify the actual ambient temperature ensures the continued validity of the ambient temperature used for the setpoint basis, and provides sufficient advance indication to take appropriate compensatory action. Accordingly, this change will not involve a significant increase in the consequences of any accident previously evaluated.

Furthermore, the LDS function provides a mitigation action for a postulated main steam line pipe leak which could lead to a pipe break. This function does not affect any accident precursors, and the proposed change does not affect the function of the LDS system. Accordingly, this change will not involve a significant increase in the probability of any accident previously evaluated.

2. The operation of NMP2 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

The qualification of safety-related equipment in the main steam lead enclosure is evaluated using actual temperatures and component qualified life is adjusted accordingly. The temperature elements are the only safety-related equipment affected by this change, therefore, the instrumentation response to previously evaluated accidents will not be adversely affected. This change will not affect the performance of safety related structures. Accordingly, the design capabilities of those structures, systems and components affected by the proposed change are not challenged in a manner not previously evaluated so as to create the possibility of a new or different kind of accident from any previously evaluated.

3. The operation of NMP2 in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

The proposed change provides a range of setpoints and allowable values for the Main Steam Line Tunnel Lead Enclosure temperatures. The calculation of the allowable values and trip setpoints was performed using the same methodologies as previously employed. For an elevated ambient temperature in the Lead Enclosure area, a setpoint established using the proposed Figure 3.3.2-1 ensures that the Main Steam Isolation Valves receive an isolation signal upon sensing a steam leak of 25 gpm, resulting in a main steam line isolation prior to a pipe break. Therefore, the proposed change provides the same level of protection against a main steam line break as the existing setpoint values. The proposed setpoints will provide increased scram avoidance, and thereby reduce unnecessary challenges to the plant shutdown systems. Accordingly, the proposed change does not result in a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Jocelyn A. Mitchell, Acting Director

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

Date of application for amendments: March 25, 1996

Description of amendment request: These amendments revise the safety limit minimum critcal power ratios (SLMCPRs) to support use of GE-13 fuel at Peach Bottom Atomic Power Station.

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

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

The derivation of the revised GE13 SLMCPRs for incorporation into the TS, and its use to determine cycle-specific thermal limits, have been performed using USNRC [U.S. Nuclear Regulatory Commission]-approved methods within the existing fuel licensing criteria as discussed in NEDE-32198P, ''GE13 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II),'' and cannot increase the probability or severity of an accident.

The basis of the SLMCPRs calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid boiling transition if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling and fuel damage in the event of a postulated accident. The fuel licensing acceptance criteria for the SLMCPRs calculation apply to the GE13 fuel in the same manner that they have applied to previous fuel designs. The probability of fuel damage is not increased. Therefore, the proposed TS changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

The SLMCPR for the GE13 fuel design is a Technical Specification numerical value, designed to ensure that transition boiling does not occur in 99.9% of all fuel rods in the core during the limiting postulated accident. It cannot create the possibility of any new type of accident. The new SLMCPRs are calculated using USNRC-approved methods and have the same calculational basis as the SLMCPR for other GE fuel designs previously used at PBAPS, Units 2 and 3. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident, from any accident previously evaluated.

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

The margin of safety as defined in the TS Bases will remain the same. The new SLMCPRs are calculated using USNRC-approved methods which are in accordance with the current fuel licensing criteria. The SLMCPRs for the GE13 fuel remain high enough to ensure that greater than 99.9% of all fuel rods in the core will avoid boiling transition if the limit is not violated, thereby preserving the fuel cladding integrity. Therefore, the proposed TS changes do not involve a reduction in a margin of safety.

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

Local Public Document Room location: Government Publications

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

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

NRC Project Director: John F. Stolz

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: June 13, 1996

Description of amendment request: The proposed amendment to the Technical Specifications (TS) will permit a one time performance of Surveillance Requirement 3.3.1.1.12, for the Average Power Range Monitor Flow Biased High Scram function, with a delayed entry into its associated TS Conditions and Required Actions for up to 6 hours provided core flow is maintained at or above 82 percent. This change would be in effect until the end of refueling outage 2R11, currently scheduled for early October 1996.

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:

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

The APRM system provides monitoring and accident mitigation functions to limit peak flux in the core during startup and run modes. This proposed TS change for delaying entry into Conditions and Required Actions associated with SR 3.3.1.1.12 for the APRM flow bias function will have no impact on the APRM system or any system that interfaces with it. No pressure boundary interfaces or process control parameters will be challenged.

This change does not affect the operation of any equipment. Delaying entry into Conditions and Required Actions associated with SR 3.3.1.1.12 does not affect either the initiator of any accident previously evaluated or any equipment required to mitigate the consequences of an accident, or the isotopic inventory in the fuel. Thus, the change does not increase either the probability or the consequences of accidents previously evaluated.

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

Because there is no direct pressure boundary interface or process control function associated with the APRM system or its interfacing electronics, the possibility of a new or different type of accident than any previously evaluated will not be created. Although the flow bias instrument loop does employ flow transmitters to measure recirculation drive flow, delaying entry into Conditions and Required Actions associated with SR 3.3.1.1.12 will have no impact on their pressure boundary function. Also, failure of the sensing line associated with these transmitters has already been accounted for in the initial plant design by including excess flow check valves for sensing line break isolation.

The proposed change does not introduce a new mode of plant operation and does not involve the installation of any new equipment or modifications to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The APRM flow biased high scram function is not specifically credited in the safety analysis. However, it is intended to provide an additional margin of protection from transient induced fuel damage during operation where recirculation flow is reduced to below the minimum required for rated power operation.

The margin of safety associated with this change refers to the margin inherent in the accident analyses that takes credit for the clamped high flux scram only (i.e., margin between scramming at 120% peak flux and the peak flux necessary for fuel damage). The current reactor operating state (end of cycle coast down extended core flow) dictates that only the 120% flux trip be enforced. This trip remains functional during the APRM flow biased high scram calibration.

Currently, the Conditions and Required Actions associated with SR 3.3.1.1.12 permit a one hour delay prior to entry because it minimizes risk while allowing time for restoration or tripping of channels by operations personnel. Because the APRM flow biased function is not enforced during end of cycle, coast down, extended core flow conditions, extending entry in associated Conditions and Required Actions from one to six hours has no impact on the margin associated with the clamped high flux scram. In the event core flow drops below 82%, the flow point below which APRM setpoints automatically become flow biased, the associated Conditions and Required Actions will be entered.

Therefore, extending entry into associated Conditions and Required Actions associated with SR 3.3.1.1.12, provided core flow remains at or above 82%, from one to six hours does not reduce 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 amendment request involves no significant hazards consideration.

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

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

NRC Project Director: John F. Stolz

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

Date of amendment request: May 16, 1996

Description of amendment request: The proposed amendment to the James A. FitzPatrick Technical Specifications (TSs) proposes to delete the requirement for the Plant Operating Review Committee (PORC) to review the fire protection program and implementing procedures. This proposal will reduce the administrative burden on the committee while making PORC's responsibilities more consistent with the other responsibilities described in Section 6.1.5.6 of the TS.

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

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

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

The proposed changes delete the Plant Operating Review Committee (PORC) review of the fire protection program and implementing procedures, and deleted fire protection inspection and audit requirements that are redundant to those performed under the cognizance of the Safety Review Committee (SRC). The changes do not introduce any new modes of plant operation, make any physical changes, or alter any operational setpoints. Therefore, the changes do not degrade the performance of any safety system assumed to function in the accident analysis. Consequently, there is no effect on the probability or consequences of an accident.

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

No physical changes to the plant or changes to equipment operating procedures are proposed. The changes are administrative and will not have any direct affect on equipment important to safety. Therefore the changes cannot create the possibility of a new or different kind of accident.

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

Adequacy of the fire protection program and implementing procedures is assured by the fire protection license condition, the procedure review and approval process implemented by Amendment 222, the provisions of 10 CFR 50.59, and inspections and audits performed under the cognizance of the SRC. Consequently, deleting PORC's responsibility for review of the fire protection program and implementing procedures, and deleting the inspection and audit requirements contained in Specification 6.14.A and 6.14.B will not degrade the fire protection program. 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, New York 10019

NRC Project Director: Jocelyn A. Mitchell, Acting Director

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

Date of amendment request: May 30, 1996

Description of amendment request: The proposed amendment would revise Minimum Critical Power Ratio Safety Limit and associated basis. The changes are required to support introduction of General Electric Company supplied, GE12, 10x10 fuel into the Cycle 13 reactor core.

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

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

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

A change in the SLMCPR [Safety Limit Minimum Critical Power Ratio] does not affect initiation of any accident. Operation in accordance with the revised SLMCPR ensures the consequences of previously analyzed accidents are not changed.

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

The SLMCPR establishes a performance limit for the fuel. Therefore changing the limit will not initiate any accident.

3. Involve a significant margin of safety because:

The analyses performed to determine the revised SLMCPR assure maintenance of the same margin of safety as presently exists for the prevention of onset of transition boiling.

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

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

Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, New York 10019

NRC Project Director: Jocelyn A. Mitchell, Acting Director

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

Date of amendment request: May 30, 1996

Description of amendment request: The proposed amendment would revise Anticipated Transient Without Scram (ATWS) Recirculation Pump Trip Reactor Pressure - High setpoint when either zero or one Safety Relief Valves are out-of service.

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

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

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

A change in the ATWS Recirculation Pump Trip Reactor Pressure - High setpoint does not affect initiation of any accident. Operation in accordance with the revised setpoints ensures the consequences of previously analyzed accidents are not changed.

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

RPV [reactor pressure vessel] pressure following an ATWS with MSIV [main steam isolation valve] closure event (worst case transient for RPV pressurization) remains within acceptable limits with the revised setpoint. Therefore changing the setpoint will not lead to a new type of accident.

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

The analyses performed to determine the revised ATWS Recirculation Pump Trip Reactor Pressure - High setpoint assure maintenance of the same margin of safety as presently exists for limiting RPV pressure following an ATWS with MSIV closure (limiting transient).

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

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

Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, New York 10019

NRC Project Director: Jocelyn A. Mitchell, Acting Director

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

Date of amendment request: May 30, 1996

Description of amendment request: The proposed amendment would eliminate selected response time testing requirements. The affected Technical Specifications (TS) are TS 4.1.A, ''Surveillance Requirements, Reactor Protection System,'' and TS 4.2.A, ''Surveillance Requirements, Instrumentation, Primary Containment Isolation Functions.''

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

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

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

The purpose of the proposed TS change is to eliminate response time testing requirements for selected sensors in the RPS [reactor protection system] and Primary Containment Isolation System. The BWROG [Boiling Water Reactor Owners Group] has completed an evaluation which demonstrates that response time testing is redundant to the other TS required testing. These other tests in conjunction with actions taken in response to NRC Bulletin 90-01, ''Loss of Fill-Oil in Transmitters

Manufactured by Rosemount,'' and Supplement 1 to Bulletin 90-01, are sufficient to identify failure modes or degradation in instrument response time and ensure operation of the associated systems within acceptable limits. Furthermore, failure modes detected by response time testing are detectable by other TS required testing. This evaluation was documented in Reference 1 [See application dated May 30, 1996]. NYPA [New York Power Authority] has confirmed the applicability of this evaluation to the FitzPatrick Plant. In addition, NYPA will complete the actions identified in the NRC staff's safety evaluation of NEDO-32291-A.

Because of the continued application of other existing TS required tests 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 and required for successful mitigation of an initiating event. The proposed changes do not affect the capability of the associated systems to perform their intended function within their required response time, nor do the proposed changes themselves affect the operation of any equipment. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not affect the ability of the systems to perform their intended function within the acceptance limits assumed in plant safety analyses and required for successful mitigation of an initiating event. No new failure modes are introduced by the changes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The current TS required response time test limits are based on the maximum allowable values assumed in the plant safety analyses. These analyses conservatively establish the margin of safety. As described above, the proposed 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 analysis. Plant and system response to an initiating event will remain in compliance within the assumptions of the safety analyses, and therefore the margin of safety is not affected.

Further, although not explicitly evaluated, the proposed changes will provide an improvement to plant safety and operation by reducing the time safety systems are unavailable, reducing safety systems actuations, reducing plant shutdown risk, limiting radiation exposure to plant personnel, and eliminating the diversion of key personnel to conduct unnecessary testing. Therefore, the overall effect of the changes should increase the margin the safety.

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

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

Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, New York 10019

NRC Project Director: Jocelyn A. Mitchell, Acting Director

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: March 6, 1996, as supplemented by letter dated May 30, 1996

Description of amendment request: The proposed change to Hope Creek Technical Specification (TS) 3.8.1, ''A.C. Sources - Operating'', would decrease the minimum fuel oil storage capacity of the Emergency Diesel Generator Fuel Oil Storage Tanks, from 48,800 to 44,800 gallons. In addition, footnote ** is deleted from TS 3.8.1.1.b.2. The proposed change would also add an Action Statement to address remedial action when a fuel oil transfer pump becomes inoperable.

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

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

TANK LEVEL

Amendment 59 provides an allowance for transferring fuel oil from a pair of storage tanks associated with an inoperable [Emergency Diesel Generator] EDG to another pair of storage tanks in order to demonstrate compliance with PSE&G's commitment to Regulatory Guide 1.137. The proposed change is consistent with that transfer strategy and extends this allowance to include using fuel oil in operable EDG storage tanks in order to reduce the amount of stored fuel oil. Transfer from operable EDG storage tanks is, actually, less complex than transferring from an inoperable EDG storage tank since power to the transfer pumps would be available.

The low level alarm setpoint is the only physical change to be made. No change is being made to the EDGs, to the fuel oil storage tanks, or to the fuel oil transfer system and since EDG fuel oil supply is associated with mitigating the consequences of an accident, there is no change in the probability of any accident analyzed in the [Updated Final Safety Analysis Report] UFSAR.

Since the proposed change still ensures the minimum fuel oil storage capacity meets the existing licensing basis and since off-site replacement oil is expected to be available within 60 hours there is no change in the consequences of an accident previously evaluated.

TRANSFER PUMP ACTION STATEMENT

Since no change is being made to the EDGs, to the fuel oil storage tanks or to the fuel oil transfer system, and since EDG fuel oil supply is associated with mitigating the consequences of an accident, there is no change in the probability of any accident analyzed in the UFSAR.

The proposed change provides compensatory action in the event a single fuel oil transfer pump is inoperable without having to immediately declare the EDG inoperable. The change ensures the affected EDG remains fully capable of functioning as assumed in the safety analyses, therefore, there is no significant impact on the consequences of an accident previously evaluated.

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

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

TANK LEVEL AND TRANSFER PUMP ACTION STATEMENT

The proposed changes will result in a setpoint change to the low level alarm. No other physical changes to the EDGs, to the fuel oil storage tanks, or to the fuel oil transfer system will result from the proposed changes. Operation including the proposed changes will not impair the diesel generators from performing as provided in the design basis. In addition, EDG fuel oil supply is associated with mitigating accident consequences, not accident prevention. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

TANK LEVEL

The margin of safety is provided by the on-site storage of an adequate supply of diesel fuel oil to ensure uninterrupted EDG operation for seven days. Although the proposed change may result in a reduction of stored fuel oil, the new minimum continues to provide for an on-site seven day supply of diesel fuel oil.

TRANSFER PUMP ACTION STATEMENT

The margin of safety is provided by the ability of the fuel oil transfer pumps to supply an adequate flow of the stored fuel to each EDG day tank. The proposed change continues to provide 100% capacity to the EDG day tank for a minimum of three days with no operator action. With the proposed action, adequate transfer capability is provided for a minimum of seven days fuel oil supply at which time refilling of the tanks would provide an indefinite supply. With both transfer pumps on a single EDG inoperable, the remaining three EDGs would provide adequate power for safe shutdown. Transfer of fuel oil from the storage tanks with inoperable transfer pumps can still be effected using temporary hoses.

Since the proposed changes do not involve the addition of plant equipment, are consistent with the intent of the existing Technical Specifications, are consistent with allowances for fuel oil transfers approved in Amendment 59, meets the intent of Regulatory Guide 1.137, and are consistent with the design basis of the diesel generators and the accident analysis, no action proposed by this request will occur that will 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: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

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

NRC Project Director: John F. Stolz

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

Date of amendment request: May 10, 1996

Description of amendment request: The proposed amendments would change Technical Specification Sections, 1.0, 2.0, 3/4 1.0, 3/4 2.0, 5.0 and 6.0. These changes support the Margin Recovery Program (MRP) and support increased steam generator tube plugging, improved fuel reliability, reduced fuel costs, longer fuel cycles, reduced spent fuel storage, and enhanced reactor safety. These changes incorporate the results of the revised safety analyses (margin recovery) and the establishment of a Core Operating Limits Report.

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

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

The accidents potentially affected by the parameters and assumptions associated with the MRP have been evaluated/ analyzed and all design standards and applicable safety criteria are met. The consideration of these changes does not result in a situation where the design, material, or construction standards that were applicable prior to the change have been altered. Therefore, the changes occurring with the MRP will not result in any additional challenges to plant equipment that could increase the probability of any previously evaluated accident.

The changes associated with the MRP do not affect plant systems such that their function in the control of radiological consequences is adversely affected. The safety evaluation documents that the design standards and applicable safety criteria limits continue to be met and therefore fission barrier integrity is not challenged. The MRP changes have been shown not to adversely affect the response of the plant to postulated accident scenarios. In all cases, the calculated doses are within the regulatory criteria and therefore do not constitute an increase in consequences. These changes will, therefore, not affect the mitigation of the radiological consequences of any accident described in the Updated Final Safety Analysis Report (UFSAR).

Based on the above, it is concluded that the probability or consequences of an accident previously evaluated is not significantly increased by the proposed changes.

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

The possibility for a new or difference[t] type of accident from any accident previously evaluated is not created since the changes associated with the MRP do not result in a change to the design basis of any plant component or system. The evaluation of the effects of the MRP changes shows that all design standards and applicable safety criteria limits are met. These changes therefore do not cause the initiation of a new accident nor create any new failure mechanisms. Component integrity is not challenged. The changes do not result in any event previously deemed incredible being made credible. The MRP changes will not result in more adverse conditions and will not result in any increase in the challenges to safety systems.

Therefore, the consideration of the MRP as described in the safety evaluation does 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 margin of safety is maintained by assuring compliance with acceptance limits reviewed and approved by the NRC. Since all of the appropriate acceptance criteria for the various analyses and evaluations have been met, by definition there has not been a reduction in any margin of safety.

Therefore, the margin of safety as defined in the Bases to the Salem Unit 1 and 2 Technical Specifications has not been significantly reduced.

Based on the above, PSE&G has determined that the proposed changes do not involve a significant hazards consideration.

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

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

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

NRC Project Director: John F. Stolz

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

Date of amendment request: April 16, 1996

Description of amendment request: The proposed amendment would revise the Virgil C. Summer Nuclear Station, Unit 1 (VCSNS), Technical Specifications (TS) to implement the amended regulation to 10 CFR Part 50, Appendix J, Option B (new rule), to provide a performance-based option for leakage-rate testing of containment. The proposed amendment will revise the VCSNS TS 3/4.6 ''Containment Systems,'' TS Bases 3/4.6, and TS 6.8 ''Administrative Controls - Programs and Procedures,'' to adopt the implementation requirements of 10 CFR Part 50, Appendix J, Option B. The proposed amendment utilizes the guidelines (guidelines) provided in ''Option B'' of Regulatory Guide (RG) 1.163 ''Performance-Based Containment Leak-Test Program, September 1995,'' and NEI 94-01, ''Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J, July 26, 1995.'' The licensee has stated that the proposed amendment is within these prescribed guidelines and does not propose any deviations to the established methods which would impact already approved analyses/ justifications and established review process.

The proposed change will remove the prescriptive TS requirements for the performance of containment leakage testing and allow leakage testing to be conducted as determined appropriate through the performance-based or risk-based alternatives described in the VCSNS Containment Leakage Rate Testing Program developed in accordance with RG 1.163 and NEI 94-01. Since the requirements of Appendix J to 10 CFR Part 50 will continue to apply, the type of testing will not change. The proposed request does not modify any plant equipment or systems.

The requirements of Appendix J will continue to govern the type of test, testing methodology, and acceptance criteria for Type A, B, and C testing. The performance-based testing of Option B eliminates or modifies prescriptive regulatory requirements for which the burden is marginal to safety for which the reviews and analyses have been presented in NUREG-1493, ''Performance-Based Containment Leak-Test Program, Final Report, September 1995.''

Earlier leakage testing performed at VCSNS has demonstrated low overall containment leakage and supports the implementation of Option 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, which is presented below:

1. The probability or consequences of an accident previously evaluated is not significantly increased.

There is no increase in the probability of an accident since there is no work that would affect containment integrity. The testing of containment isolation valves (CIVs) and other containment penetration sealing devices is not postulated as an accident precursor or initiating event.

Type A testing is capable of determining the total leakage from both local leakage paths and gross containment leakage paths. Our Type B and C testing has consistently provided accurate leakage rates for valves and penetrations.

Administrative controls govern maintenance and testing such that there is very low probability that unacceptable maintenance or alignments can occur. Prior to and following maintenance on CIVs and penetrations, a local leak rate test (LLRT) is required to be performed. As a result, Type A testing is not required to accurately quantify the leakage through containment penetrations.

Any specific exemptions to the requirements of Appendix J will require approval by the NRC before implementation.

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

2. The possibility of an accident or a malfunction of a different type than any previously evaluated is not created.

The proposed request does not involve any physical changes to the plant, affect the operation of the plant, or change testing methods or acceptance criteria. The history of containment testing verifies that containment integrity has been maintained.

The frequency changes allowed by implementation of Option B will not significantly decrease the level of confidence in the ability of the reactor building to limit offsite doses to allowable values. No accident or malfunction can be the result of the allowed changes to test schedule or frequency.

Since the proposed request will not directly impact equipment, procedures or operations, the changes will not create the possibility of any new or different kind of accident from any previously evaluated.

3. The margin of safety has not been significantly reduced.

The reason for performing containment leakage rate testing is to assure that the leakage paths are identified, and that any accident release will be restricted to those paths assumed in the safety analysis. The purpose for the schedule is to assure that containment integrity is verified on a periodic basis.

Implementation of Option B to provide flexibility in the scheduled requirements does not mean that containment integrity will be compromised. The historical leakage rate test results for VCSNS and for the nuclear industry support extension of testing frequencies and demonstrate that structural integrity has been maintained.

Therefore, the margin of safety has not been significantly reduced.

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: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180

Attorney for licensee: Randolph R. Mahan, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, SC 29218

NRC Project Director: Eugene V. Imbro

Southern Nuclear Operating Company, Inc., Docket No. 50-364, Joseph M. Farley Nuclear Plant, Unit 2, Houston County, Alabama

Date of amendment request: April 22, 1996

Description of amendment request: The amendment would revise the Technical Specifications to implement the L* Tubesheet Region Plugging Criterion, which would allow a steam generator tube to remain in service with bands of axial degradation in the tubesheet region provided sufficient non-degraded tubing remains to satisfy regulatory guidance concerning structural and leakage integrity.

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 Farley Nuclear Plant Unit 2 steam generators in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The supporting technical evaluations of the subject criteria demonstrate that the presence of the tubesheet enhances the tube integrity in the region of the hardroll by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and tube collapse is strengthened by the presence of the tubesheet in that region. The result of the hardroll of the tube into the tubesheet is an interference fit between the tube and the tubesheet. Tube rupture [cannot] occur because the contact between the tube and tubesheet does not permit sufficient movement of tube material. In a similar manner, the tubesheet does not permit sufficient movement of tube material to permit buckling collapse of the tube during postulated LOCA [loss-of-coolant accident] loadings.

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

The SRE [sound roll expansion] L* length is sufficient to preclude significant leakage from tube degradation located below the L* length. The existing Technical Specification leak rate requirements and accident analysis assumptions remain unchanged in the unlikely event that significant leakage from this region does occur. Any leakage from the tube within the tubesheet at any elevation in the tubesheet is fully bounded by the existing steam generator tube rupture analysis included in the Farley Nuclear Plant Final Safety Analysis Report. A conservative leakage allowance for each L* tube is provided to determine the impact of L* criterion upon offsite doses in the event of a postulated double ended guillotine break of the main steam line outside of containment, but upstream of the main steam line isolation valves. Since Farley Unit 2 has implemented the Interim Plugging Criteria (IPC) for ODSCC at the tube support plates, projected steam line break (SLB) leakage at the end of the next successive operating cycle must be evaluated. Per Generic Letter 95-05, plants implementing the IPC can utilize SLB leakage limits higher than the originally assumed 1.0 gpm primary to secondary leakage value provided an analysis of offsite doses consistent with the Standard Review Plan methodology is performed. This analysis performed for the Farley Unit 2 plant indicates that primary to secondary leakage of 11.2 gpm in the faulted loop (0.1 gpm in the intact loops) will result in offsite doses at the site boundary of less than 10% of the 10 CFR [Part] 100 guidelines. The total projected SLB leakage from all leakage sources must remain below this value. Per attachment 4 addressing the L* methodology, the number of tube ends to which L* criterion can be applied is limited to 600 per steam generator. Using a bounding SLB leakage allowance per L* tube, the SLB leakage component from 600 L* tube ends will be less than 0.33 gpm in the faulted loop. The proposed alternate plugging criterion does not adversely impact any other previously evaluated design basis accident. As the current Unit 2 IPC SLB leakage has been calculated to be less than 2 gpm in the faulted loop, [an] SLB leakage margin of over 9 gpm is provided for this cycle.

As noted above, tube rupture and pullout is not expected for tubes using the L* criterion. In addition to the L* length, a minimum length of SRE below the identified degradation must be established. The aggregate L* distance of SRE provides the structural integrity to prevent tube pullout. Conservatively, it is assumed that the degraded band length does not provide any support in resisting tube pullout.

Therefore SNC [Southern Nuclear Operating Company, Inc.] concludes that Operation of the Farley Nuclear Plant Unit 2 steam generators in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Implementation of the proposed L* criterion does not introduce any significant changes to the plant design basis. Use of the criterion does not provide a mechanism to result in an accident initiated outside of the region of the tubesheet expansion. The structural integrity of L* tubes will be maintained during all plant conditions. Any hypothetical accident as a result of any tube degradation in the expanded portion of the tube would be bounded by the existing tube rupture accident analysis. If it is postulated that a circumferential separation of an L* tube were to occur below the PLRL [pullout load reaction length], tube structural and leakage integrity will be maintained during all plant conditions. Verification of the L* distance of non-degraded tube roll expansion prevents the postulated separated tube from lifting out of the tubesheet during all plant conditions. Verification of the L* criterion prevents tube displacement of any magnitude, and therefore, postulated axial cracks existing a minimum of 0.5 inch from either the bottom of the roll transition or top of tubesheet, whichever is lower, from migrating out of the tubesheet.

Therefore, SNC concludes that the proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The use of the L* criterion has been concluded to maintain the integrity of the tube bundle commensurate with the requirements of draft Regulatory Guide 1.121 under normal and postulated accident conditions. The safety factors used in the verification of the strength of the degraded tube are consistent with the safety factors in the ASME [American Society of Mechanical Engineers] Boiler and Pressure Vessel Code used in steam generator design. The L* length has been verified by testing to be greater than the length of roll expansion required to preclude significant leakage during normal and postulated accident conditions. The leak testing acceptance criteria are based on the primary to secondary leakage limit in the Technical Specifications and the leakage assumptions used in the FSAR accident analyses. The L* distance provides for structural integrity during all plant conditions.

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

Therefore, SNC, concludes based on the above, it is concluded that the proposed change does not result in a significant reduction in a loss of margin with respect to plant safety as defined in the Final Safety Analysis Report [FSAR] or the bases of the FNP [Farley Nuclear Plant] 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: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

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NRC Project Director: Herbert N. Berkow

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

Date of application request: May 29, 1996

Description of amendment request: The application requests staff review and approval of a modification to the facility, as described in the safety analysis report, that involves an unreviewed safety question. The modification will reduce the single failure trip potential for the main feedwater control and bypass 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. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Callaway safety analysis assumes the MFC&BVs [main feedwater control and bypass valves] close during certain events in order to terminate fluid inventory addition to faulted steam generators and thereby preclude the diversion of auxiliary feedwater to the main feedwater system. This feature is necessary because each feedwater line at Callaway is equipped with only one MFIV [main feedwater isolation valve]. It should be noted that the safety analysis simply requires the valves to close and does not prescribe a mechanism for accomplishing that action.

The following are accidents that credit feedwater isolation or AFW [auxiliary feedwater] addition. There is no impact by the proposed modification on the consequences of each accident.

- Feedwater System Malfunctions That Result In An Increase In Feedwater Flow
- Inadvertent Opening Of A Steam Generator Relief or Safety Valve
- Steam System Piping Failure
- Loss of Nonemergency AC Power to the Station Auxiliaries
- Loss of Normal Feedwater Flow
- Feedwater System Pipe Break
- Decrease in Reactor Coolant Inventory

The modification will not change the radiological consequences of FSAR [final safety analysis report] Chapter 15 accidents because the feedwater isolation function (and NSSS [nuclear steam supply system] break response) has not changed. Therefore, there will be no increase in the consequences of an accident evaluated previously in the FSAR.

An analysis was performed to quantify the impact of the proposed modification on the probability of MFCV [main feedwater control valve] failure (closure) during normal plant operation. Comparison of this failure probability for the existing design (1.20E-1 per year) versus the proposed design (6.99E-2 per year) indicates that the percentage reduction in the system failure probability at power is 41.75%. Thus, the proposed design results in a reduction in the probability of inadvertent MFCV failures at power and hence, a reduction in the probability of a reactor trip and subsequent challenges to other safety systems.

While this modification reduces the probability of a reactor trip, it slightly increases the unavailability of the feedwater isolation function. This is because the original design required actuation of only one FWIS [feedwater isolation system] train to close the MFC&BVs, whereas the new design requires actuation of both trains. The impact of the modification on the probability of incurring a feedwater isolation failure was therefore quantified, utilizing PRA [probabilistic risk assessment] techniques. Fault trees were developed for both the new and existing designs. Failure probabilities for each event were then obtained from the IPE [individual plant examination] and utilized to calculate failure probabilities for the feedwater isolation safety function. This calculation considered hardware failures only, i.e., failure of an MFIV to close after receiving an actuation signal. The failure probability of feedwater isolation, based on the proposed design, was determined to be 6.1E-5 per demand (1 event every 16,400 demands). The existing design was found to have a failure probability of 2.8E-5 per demand (1 event every 35,700 demands). Therefore, this modification will not significantly increase the probability or consequences of an accident evaluated previously in the FSAR.

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

The modification maintains the present de-energize-to-actuate configuration of the MFC&BV trip solenoid valves.

Thus, the proposed modification does not create the possibility of an accident of a different type than any previously evaluated.

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

Credit is taken in the accident analyses for the MFIVs to close on demand for feedwater isolation. Because of this, the MFIVs have been incorporated into the Callaway Technical Specifications. Action Statements and surveillance requirements have been developed to assure the availability of the valves when needed.

The MFC&BVs are not addressed by any of the Callaway Technical Specifications or their bases. Therefore, this modification will not involve a significant reduction in the margin of safety as defined in the basis for any technical specification.

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

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

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

NRC Project Director: William H. Bateman

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

Date of amendment request: May 29, 1996

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 15.4.4, ''Containment Tests,'' to incorporate the provisions of 10 CFR Part 50, Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,'' Option B. Revisions would also be made to TS Sections 15.1, ''Definitions,'' 15.3.6, ''Containment System,'' and 15.6, ''Administrative Controls,'' to support the proposed changes to Section 15.4.4.

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 this facility under the proposed Technical Specifications will not create 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 or consequences of an accident previously evaluated in the PBNP [Point Beach Nuclear Plant] Final Safety Analyses Report (FSAR). Furthermore, containment leakage rate testing is not an initiator of any accident. The proposed change simply provides a mechanism within the Technical Specifications for implementing a performance-based method of determining the frequency for leakage rate testing which has been approved by the NRC. The proposed change does not affect reactor operations or accident analysis and has no significant radiological consequences. Therefore, this change will not create a significant increase in the probability or consequences of an accident previously evaluated.

2. Operation of this facility under the proposed Technical Specifications change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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. This change involves a potential reduction of Type A, B, and C test frequency. Except for the method of defining the test frequency, the methods for performing the actual tests are 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. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of this facility under the proposed Technical Specifications change will not create a significant reduction in a margin of safety.

The proposed change potentially affects only the frequency of Type A, B, and C testing. Except for the method of defining test frequency, the methods for performing the actual tests are not changed. The proposed change is based on NRC accepted provisions and maintains necessary levels of system and component reliability affecting containment integrity. Evaluation of the performance-based approach to leakage rate testing, as documented in NUREG-1493, concludes that the impact on public health and safety due to revised testing intervals is negligible. Furthermore, the proposed change will not reduce the availability of systems associated with containment integrity when they are required to mitigate accident conditions. Therefore, the proposed change will not create 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: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: June 4, 1996

Description of amendment request: The proposed amendment would revise the Kewaunee Nuclear Power Plant Technical Specifications (TS) by reducing the surveillance test frequencies for the radiation monitoring system (Table TS 4.1-1) and the control rods (Table TS 4.1-3) in accordance with the guidance of Generic Letter 93-05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation,'' dated September 27, 1993.

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:

Table TS 4.1-1, ''Minimum Frequencies for Checks, Calibrations and Test of Instrument Channels,'' Item 19

The proposed changes were reviewed in accordance with the provisions of 10 CFR 50.92 to determine that no significant hazards exist. The proposed changes will not:

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

The radiation monitors are not accident initiators; therefore, they cannot increase the probability of an accident occurring. The reliability of the radiation monitors is not expected to decrease due to the decreased surveillance frequency; therefore, this change does not increase the consequences of an accident.

The addition of comment (a) to the Check, Calibrate, and Test columns is merely a clarification of the existing information in the table and does not change the intent of the Technical Specifications.

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

The proposed change revises only the testing frequency and does not revise the test method or operational performance of the radiation monitors. The radiation monitors are not accident initiators; therefore, they cannot create a new or different kind of accident.

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

Quarterly testing of the radiation monitoring system channels will continue to verify operability of the monitors. Decreasing the test surveillance frequency is not expected to decrease the reliability of the radiation monitors. This change is acceptable in accordance with Generic Letter 93-05 and NUREG-1366, ''Improvements to Technical Specifications Surveillance Requirements.''

Table TS 4.1-3, ''Minimum Frequencies for Equipment Tests,'' Item 1

The proposed change in test frequency for control rod exercising was reviewed in accordance with the provisions of 10 CFR 50.92 to determine that no significant hazards exist. It has been determined that the proposed change will not:

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

The proposed change revises only the testing frequency for control rod exercising. The control rod exercise surveillance procedure will continue to be conducted, on a quarterly basis, to ensure that the equipment remains operable. The reduced frequency of control rod exercising reduces the probability of an inadvertent reactor trip occurring during testing due to a dropped control rod. Surveillance procedure SP 49-075 is conducted to verify rod movement. In accordance with NUREG-1366, the frequency of a stuck control rod occurring is very low. This condition is most often discovered during reactor startup or during low power physics testing. The reduction in control rod exercising is, therefore, considered acceptable and is not expected to affect the probability of a stuck control rod occurring.

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

The proposed change revises only the testing frequency and does not revise the test method or the design of the control rod system. Therefore, a new or different kind of accident will not be created by this change.

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

Quarterly control rod exercising will continue to verify movement of the control rods. No adverse consequences are expected to occur due to decreasing the test frequency. This change is acceptable in accordance with Generic Letter 93-05 and NUREG-1366.

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

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, Wisconsin 54311-7001

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P. O. Box 1497, Madison, Wisconsin 53701-1497

NRC Project Director: Gail H. Marcus

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

Date of amendment request: June 10, 1996

Description of amendment request: The proposed amendment would revise Technical Specification 4.2.b, ''Steam Generator Tubes,'' and its associated basis, by allowing the use of Westinghouse laser-welded sleeves to repair defective steam generator tubes. A description of the sleeving repair process and supporting technical justification are contained in WCAP-13088, Revision 3, ''Westinghouse Series 44 and 51 Steam Generator Generic Sleeving Report.'' WCAP-13088, and a non-proprietary version (WCAP-13089), were submitted to the Nuclear Regulatory Commission on April 13, 1995, to support a similar TS amendment request for the DC Cook Nuclear Power Plant, Unit 1.

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

1. Operation of the KNPP [Kewaunee Nuclear Power Plant] in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The LWS [laser-welded sleeve] configuration has been designed and analyzed in accordance with the requirements of the ASME [American Society of Mechanical Engineers] Code. Fatigue and stress analyses of the sleeved tube assemblies produced acceptable results; i.e., the applied stresses and fatigue usage for the sleeve and weld are bounded by the limits established in the ASME Code. ASME Code minimum material property values are used for the structural and plugging limit analysis. Ultrasonic inspection is used to verify that minimum weld fusion zone thicknesses are produced. Mechanical testing of 7/8'' tubesheet sleeves installed in roll expanded tubes has shown that the individual joint structural strength of Alloy 690 LWSs provides margin to acceptance limits. These acceptance limits bound the most limiting loadings (3 times normal operating pressure differential) recommended by RG [Regulatory Guide] 1.121. Therefore, each individual joint provides for structural integrity exceeding RG recommendations. A hypothetical loss of integrity of one of the joints will not result in a loss of structural integrity for the sleeve. Leakage testing for 3/4'' and 7/8'' full length tubesheet sleeves has demonstrated that unacceptable levels of primary-to-secondary leakage are not expected during all plant conditions for non-welded tubesheet sleeve lower joints. The welded joint produces a hermetic seal, and therefore will not leak under any plant conditions. Laser welded sleeves will not contribute to the current SLB [steam-line break] primary-to-secondary leakage limit of 34 gpm in the faulted loop. The 34 gpm leakage limit was calculated in accordance with the standard review plan methodology to support implementation of the voltage-based repair criteria for tube support plate intersections.

The sleeve minimum acceptable wall thickness (used for developing the depth based plugging limit for the sleeve) is determined using the guidance of RG 1.121 and the pressure stress equation of Section III of the ASME Code. With respect to the design of the sleeve for KNPP, the limiting requirement of the RG which applies to part throughwall degradation is that the minimum acceptable wall must maintain a factor of safety consistent with the analysis conditions as defined by the ASME Code. A bounding set of design and transient loading input conditions was used for the minimum wall thickness evaluation in the generic evaluation. Evaluation of the minimum acceptable wall thickness for normal, upset and postulated accident condition loading per the ASME Code indicates the limiting condition is established for the normal operating conditions, and the minimum acceptable wall thickness for this case bounds the upset and faulted condition values.

According to RG recommendations, an allowance for non-destructive evaluation (NDE) uncertainty and operational growth of existing tube wall degradation indications within the sleeve must be accounted for when determining the sleeve plugging limit. A conservative tube wall degradation growth rate per cycle and an NDE uncertainty has been assumed for determining the sleeve TS plugging limit. The sleeve wall degradation extent determined by NDE, which would require plugging sleeved tubes, is developed using the guidance of RG 1.121 and is defined in WCAP-13088 [non-proprietary WCAP-13089] to be 25% throughwall (plugging limit = 100% - structural limit + NDE uncertainty + growth) for KNPP.

The hypothetical consequences of failure of the sleeve joint would be bounded by the current SG [steam generator] tube rupture analysis included in the KNPP Updated Safety Analysis Report. Due to the slight reduction in diameter caused by the sleeve wall thickness, primary coolant release rates would be slightly less than assumed for the SG tube rupture analysis (depending on break location), and therefore, would result in lower total primary fluid mass release to the secondary system.

The proposed TS change to use Alloy 690 LWSs does not adversely impact any other previously evaluated design basis accidents or the results of LOCA [loss of coolant accident] and non-LOCA accident analyses for the current TS minimum reactor coolant system flow rate. The results of the analyses and testing, as well as plant operating experience, demonstrates that the sleeve assembly is an acceptable means of maintaining tube integrity. Plugging limit criteria are established using the guidance of RG 1.121. Furthermore, per RG 1.83 recommendations, the sleeved tube will be monitored through periodic inspections with present NDE techniques. These measures demonstrate that installation of sleeves spanning degraded areas of the tube will restore the tube to a condition consistent with its original design basis.

Corrosion testing of free span LWS joint has indicated that the corrosion resistance (relative to roll transitions) can be increased by greater than a factor of ten with the application of a PWHT [post weld heat treatment] step. Estimations of joint susceptibility based on expected far field stresses after heat treatment using the expected original tube-to-tubesheet hydraulic expansion transition residual stresses and actual time to crack in these transitions at KNPP indicate that LWS joint lifetime should exceed the current plant license. Consistent with other license amendments addressing LWS, all free span laser welds will receive a PWHT; therefore, rapid corrosion degradation of the free span joint is not expected. Recently performed corrosion testing of LWS joints in locked tube conditions indicates that with PWHT the stress corrosion cracking resistance and initiation potential in the parent tube weld region is greatly enhanced. Similar test results and conclusions would be expected for KNPP. The Model 51 SG tube span between the top of the tubesheet and the first support plate is such that even lower PWHT residual stresses would be expected. Also, the weld placement within the hydraulically expanded area and sleeve installation sequence have been optimized to provide for some level of heat treatment at the upper transition above the weld and lower far field residual stress levels. While no parent tube degradation has been detected at this elevation, or any other elevation in a laser welded sleeve assembly, the relocation of the weld serves to provide further resistance to PWSCC [primary water stress corrosion cracking] at this elevation. The suggested target PWHT temperature has also been optimized in that this temperature provides for adequate PWHT while maintaining the parent tube far field stresses.

Approximately 19,500 LWSs have been installed in the U.S. Of this number, over 300 which have up to 3 cycles of operation were inspected in 1995 using the CECCO-5 probe. No degradation of the sleeves or the parent tube was detected. Operating experience in Europe has shown good performance of the LWS joint for up to 5 cycles of operation. In 1994, approximately 11,200 LWSs were installed in the Doel-4 Plant. After one year of operation, all in-service sleeves were inspected using the +point probe. No service induced corrosion was detected. In 1995, approximately 18,600 LWSs were installed in two different U.S. plants. Due to their limited operational time, these sleeves have not been inspected.

Conformance of sleeve design with the applicable sections of the ASME Code and results of the leakage and mechanical tests support the conclusion that installation of LWSs will not increase the probability or consequences of an accident previously evaluated.

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

Installation of LWSs will not introduce significant or adverse changes to the plant design basis and does not represent a potential to affect any other plant component. Stress and fatigue analysis of the repair has shown that the ASME Code and RG 1.121 criteria are not exceeded. Installation of LWSs maintains overall tube bundle structural and leakage integrity at a level consistent to that of the originally supplied tubing during all plant conditions; stresses are bounded by the Code and the tubing is leaktight. Sleeving of tubes does not provide a mechanism resulting in an accident outside of the area affected by the sleeves. Any hypothetical accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing tube rupture accident analysis. Since the sleeve design does not affect any component or location of the tube outside of the immediate area repaired, in addition to the fact that the installation of sleeves and the impact on current plugging level analyses is accounted for, the possibility that laser welded sleeving creates a new or different type of accident is not supported.

Installation of LWSs will reduce the potential for primary-to-secondary leakage during postulated steam line break while not significantly impacting primary coolant flow area in the event of a LOCA. By effectively isolating degraded areas of the tube through repair, the potential for steam line break leakage is reduced.

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

The LWS repair of degraded SG tubes as identified in WCAP-13088 [non-proprietary WCAP-13089] has been shown by analysis to restore the integrity of the tube bundle consistent with its original design basis conditions; i.e., tube/sleeve operational and faulted conditions stresses and cumulative fatigue usage are bounded by the ASME Code requirements and the repaired tubes are leaktight. The safety factors used in the design of sleeves for the repair of degraded tubes are consistent with the safety factors in the ASME Code used in SG design. The design of the LWS lower joint for 7/8'' tube sleeves has been verified by testing to sufficiently preclude leakage during normal and postulated accident conditions. The portions of the installed sleeve assembly which represents the reactor coolant pressure boundary will be monitored for the initiation and progression of sleeve/tube wall degradation, thus satisfying the requirements of RG 1.83. The portion of the tube bridged by the sleeve joints is effectively removed from the pressure boundary, and the sleeve then forms the new pressure boundary. The areas of the sleeved tube assembly which require inspection are defined in WCAP-13088 [non-proprietary WCAP-13089]. Since the installed sleeves represent a portion of the pressure boundary, a baseline inspection of these areas is required prior to operation with sleeves installed.

The effect of sleeving on the design transients and accident analyses has been reviewed based on the installation of sleeves up to the level of SG tube plugging coincident with the minimum reactor coolant flow rate. The installation of sleeves is evaluated as the equivalent of some level of SG tube plugging. This is based on determining the minimum reactor coolant flow for the LOCA evaluation. Information provided in WCAP-13088 [non-proprietary WCAP-13089] describes the method to determine the flow equivalent for all combinations of tubesheet and tube support plate sleeves. Therefore, installation of LWSs 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: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, Wisconsin 54311-7001

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P. O. Box 1497, Madison, Wisconsin 53701-1497

NRC Project Director: Gail H. Marcus

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

Date of amendment request: June 4, 1996 (VPNPD-96-035)

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 15.2.3, ''Limiting Safety System Settings and Protective Instrumentation,'' and Section 15.5.3, ''Design Features - Reactor,'' to incorporate changes associated with the operation of Point Beach Nuclear Plant (PBNP), Unit 2, with replacement steam generators.

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 this facility under the proposed Technical Specifications will not

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

The proposed changes do not involve a change to structures, systems, or components which would affect the probability or consequences of an accident previously evaluated in the PBNP Final Safety Analyses Report (FSAR). The proposed setpoints maintain the margin to safe operation of Unit 2 with the replacement steam generators. In order to maintain one set of safety analyses for both units, the analyses for operation of Unit 2 with the replacement steam generators were performed to encompass the operation of Unit 1. Therefore, the proposed changes apply to the operation of both units and maintain the margin of safety for each. The proposed change to the description of nominal RCS [reactor coolant system] volume is an administrative change and has no effect on plant operation. Therefore, the probability or consequences of a previously evaluated accident are not significantly increased as a result of these changes.

2. Operation of this facility under the proposed Technical Specifications change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not involve a change to the plant design. The proposed setpoints maintain the margin to safe operation of Unit 2 with the replacement steam generators. In order to maintain one set of safety analyses for both units, the analyses for operation of Unit 2 with the replacement steam generators were performed to encompass the operation of Unit 1. Therefore, the proposed changes apply to the operation of both units and maintain the margin of safety for each. These changes do not affect any of the parameters or conditions that contribute to initiation of any accidents. The proposed change to the description of nominal RCS volume is an administrative change and has no effect on plant operation or initiation of any accidents. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of this facility under the proposed Technical Specifications change will not create a significant reduction in a margin of safety.

The proposed setpoints maintain the margin to safe operation of Unit 2 with the replacement steam generators. In order to maintain one set of safety analyses for both units, the analyses for operation of Unit 2 with replacement steam generators were performed to encompass the operation of Unit 1. Therefore, the proposed changes apply to the operation of both units and maintain the margin of safety for each. The proposed change to the description of nominal RCS volume is an administrative change and has no effect on plant operation. Therefore, the proposed changes will not create 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: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: June 4, 1996 (VPNPD-96-036)

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 15.2.1, ''Safety Limit, Reactor Core,'' 15.2.3, ''Limiting Safety System Settings, Protective Instrumentation,'' and Section 15.3.1.G, ''Operational Limitations,'' to maintain safety margin for Unit 2 with replacement steam generators.

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 this facility under the proposed Technical Specifications will not create 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 or consequences of an accident previously evaluated in the PBNP [Point Beach Nuclear Plant] Final Safety Analyses Report (FSAR). The proposed changes maintain the margin to safe operation for Unit 2 with the replacement steam generators. In order to maintain one set of safety analyses for both units, the analyses for operation of Unit 2 with the replacement steam generators were performed to encompass the operation of Unit 1. Therefore, the proposed changes apply to the operation of both units and maintain the margin of safety for each. The proposed changes do not change, degrade, or preclude the prevention or mitigation of the consequences of any accident described in the FSAR. Therefore, the probability or consequences of a previously evaluated accident are not significantly increased as a result of these changes.

2. Operation of this facility under the proposed Technical Specifications change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not involve a change to the plant design. The proposed changes maintain the margin to safe operation for Unit 2 with the replacement steam generators. In order to maintain one set of safety analyses for both units, the analyses for operation of Unit 2 with the replacement steam generators were performed to encompass the operation of Unit 1. Therefore, the proposed changes apply to the operation of both units and maintain the margin of safety for each. These changes do not affect any of the parameters or conditions that contribute to initiation of any accidents. In addition, the safety functions of safety-related systems and components, which are related to accident mitigation, have not been altered. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of this facility under the proposed Technical Specifications change will not create a significant reduction in a margin of safety.

The proposed changes maintain the margin to safe operation for Unit 2 with the replacement steam generators. In order to maintain one set of safety analyses for both units, the analyses for operation of Unit 2 with replacement steam generators were performed to encompass the operation of Unit 1. Therefore, the proposed changes apply to the operation of both units and maintain the margin of safety for each. The proposed changes have no affect on the availability, operability, or performance of the safety-related systems and components described in the Technical Specifications. Therefore, the proposed changes will not create 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: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241

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

NRC Project Director: Gail H. Marcus

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

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

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

Florida Power and Light Company, et al., Docket No. 50-335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of amendment request: June 1, 1996

Description of amendment request: Revise Technical Specifications to reflect reduced reactor coolant system flows resulting from increased percentage of plugged steam generator tubes.

Date of publication of individual notice in the Federal Register: June 7, 1996 (61 FR 29140)

Expiration date of individual notice: June 24, 1996

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

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

Date of amendment request: May 31, 1996

Brief description of amendment request: The amendments (1) revise the Reactor Vessel Level Indication System (RVLIS) Action Statements to facilitate actions necessary for channel testing to be performed in Mode 3, (2) revise the Channel Calibration definition to better account for temperature detector channel calibration methodology, and (3) delete a requirement to install a jumper in the Auxiliary Feedwater actuation logic since a design change will result in the jumper function being performed by a relay.

Date of publication of individual notice in Federal Register: June 17, 1996 (61 FR 30641)

Expiration date of individual notice: July 17, 1996

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

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendments: May 28, 1996, as supplemented by letters dated May 31 and June 5, 1996

Brief description of amendments: These amendments authorize the licensee to revise applicable Updated Final Safety Analysis Report sections to reflect the installation of a variable flow controller for the service water inlet control valves for the containment air coolers that is not within the current licensing basis of Calvert Cliffs Nuclear Power Plant Units No. 1 and No. 2. These amendments are being issued pursuant to the requirements of 10 CFR 50.59(c) because the review by Baltimore Gas and Electric Company identified the changes as an unreviewed safety question. No changes to the Technical Specifications are required by these amendments.

Date of issuance: June 17, 1996

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

Amendment Nos.: 215 and 192

Facility Operating License Nos. DPR-53 and DPR-69: The amendments revised the Updated Final Safety Analysis Report. Public comments requested as to proposed no significant hazards consideration: Yes (61 FR 27371 dated May 31, 1996). 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 July 1, 1996, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendments. The May 31 and June 5, 1996, letters provided additional information that did not change the scope of the May 28, 1996, application.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and a final no significant hazards consideration determination are contained in a Safety Evaluation dated June 17, 1996.

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

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

Date of application for amendment: October 24, 1994, as supplemented August 31, 1995 and February 8, 1996. The August 31, 1995 and February 8, 1996, letters provide clarification information. The new information changed the scope of the October 24, 1994, letter and was re-noticed on May 8, 1996, but did not change the initial no significant hazards consideration determination.

Brief description of amendment: The proposed amendment would revise the TS to allow the relocation of TS 3/4.11.2.6, Gas Storage Tanks; and the associated Bases in the TS to licensee-controlled documents.

Date of issuance: June 12, 1996

Effective date: June 12, 1996

Amendment No.: 64

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

Date of initial notice in Federal Register: November 23, 1994 (59 FR 60379). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated

June 12, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendments: December 21, 1995

Brief description of amendments: The amendments delete the requirement to place the reactor mode switch in the Shutdown position if a stuck open safety/relief valve can not be closed within 2 minutes. The operator will still be required to scram the reactor if suppression pool average water temperature reaches 110 degrees Fahrenheit or greater. The amendment also includes editorial changes to the index pages.

Date of issuance: June 18, 1996

Effective date: Immediately, to be implemented within 60 days

Amendment Nos.: 113 and 98

Facility Operating License Nos. NPF-11 and NPF-18: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20844) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 18, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendments: July 18, 1994, as supplemented by letter dated October 9, 1994

Brief description of amendments: The amendments revise the current combined Technical Specifications (TS) for Units 1 and 2 by separating them into individual volumes for Unit 1 and Unit 2. In addition to the changes required by the TS split, some administrative and editorial changes were made, such as the correction of typographical errors and the deletion of unnecessary blank pages.

Date of issuance: June 12, 1996

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

Amendment Nos.: 148 and 142

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

Date of initial notice in Federal Register: September 14, 1994 (59 FR 47166) The October 9, 1995 and June 6, 1996, letters provided clarifying information that did not change the scope of the July 18, 1994, application and the initial proposed no significant hazards consideration. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 12, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendment: February 22, 1996

Brief description of amendment: The amendment increased the safety function lift setpoint tolerances for the safety and relief valves that are listed in Surveillance Requirement 3.4.4.1 (Page 3.4-10) of the Technical Specifications (TSs) for the Grand Gulf Nuclear Station, Unit 1. The tolerances were increased from the current plus or minus 1 percent of the safety function (i.e., safety relief valve) lift setpoint to plus or minus 3 percent.

Date of issuance: June 12, 1996

Effective date: June 12, 1996

Amendment No: 123

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

Date of initial notice in Federal Register: March 27, 1996 (61 FR 13524) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 12, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendment: May 1, 1996

Brief description of amendment: The amendment revises the Operating License and Technical Specifications (TS) to implement 10 CFR Part 50, Appendix J - Option B, by referring to Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program.'' Specifically, changes have been made to paragraph 2.D of the Operating License; TS Section 1.1, ''Definitions;'' TS 3.6.1.1, ''Primary Containment;'' TS 3.6.1.1, ''Primary Containment Air Locks;'' TS 3.6.1.3, ''Primary Containment Isolation Valves (PCIVs);'' and TS Section 5.5, ''Programs and Manuals.''

Date of issuance: June 21, 1996

Effective date: June 21, 1996

Amendment No.: 105

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

Date of initial notice in Federal Register: May 21, 1996 (61 FR 25708) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 21, 1996. No significant hazards consideration comments received: No

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

Niagara Mohawk Power Corporation, Docket No. 50-410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: January 25, 1996

Brief description of amendment: The amendment revises Technical Specification 3/4.3.3, Emergency Core Cooling System Actuation Instrumentation, to more clearly define when, during shutdown and refueling, the Loss of Voltage and Degraded Voltage relays for the Loss of Power actuation trip functions are required to be operable.

Date of issuance: June 10, 1996

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

Amendment No.: 72

Facility Operating License No. NPF-69: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20851) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 10, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendment: January 5, 1996, as supplemented on May 31, 1996

Brief description of amendment: The amendment implements the guidance of Generic Letter 93-08 by relocating Tables 3.3-2, ''Reactor Protective Instrumentation Response Times'' and 3.3-5, ''Engineered Safety Features Response Times'' from the Technical Specifications to the Millstone Unit No. 2 Technical Requirements Manual (TRM). In accordance with Generic Letter 93-08, the Limiting Conditions for Operations for Technical Specifications 3.3.1.1, 3.3.2.1, and 3.7.1.6 are revised to eliminate their references to the aforementioned tables. The amendment also revises Bases 3/4.3.1 and 3/4.3.2 to reference that the instrument response times are located in the TRM and that these tables in the TRM are now controlled under 10 CFR 50.59. The amendment also removes a cycle-specific note from Tables 3.3-3 and 3.3-4.

Date of issuance: June 10, 1996

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

Amendment No.: 198

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

Date of initial notice in Federal Register: February 14, 1996 (61 FR 5816) The May 31, 1996, letter provided additional information that did not change the scope of the January 5, 1996, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 10, 1996. No significant hazards consideration comments received: No

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360 and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut 06385

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: February 15, 1996

Brief description of amendments: The amendments change the Technical Specifications to implement 10 CFR Part 50, Appendix J, Option B, by creating Technical Specification Section 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' which refers to Regulatory Guide 1.163, ''Performance-Based Containment Leakage-Test Program.''

Date of issuance: June 18, 1996

Effective date: Both units, as of date of issuance, to be implemented by June 28, 1996.

Amendments Nos.: 214 and 219

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

Date of initial notice in Federal Register: March 27, 1996 (61 FR 13531) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 18, 1996. 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

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: April 25, 1996

Brief description of amendments: The amendments relocate Technical Specification Traversing In-Core Probe System Limiting Condition for Operation 3/4.3.7.7 and its Bases 3/4.3.7.7, to the Limerick Generating Station Technical Requirements Manual, and modify Note (f) of TS Table 4.3.1.1-1.

Date of issuance: June 11, 1996

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

Amendment Nos.: 117 and 79

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

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20840) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 11, 1996 No significant hazards consideration comments received: No

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

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

Date of application for amendment: March 14, 1996

Brief description of amendment: The proposed changes would allow a one-time extension of the intervals for the steam generator tube inspection that is due in July 1996.

Date of issuance: June 19, 1996

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

Amendment No.: 166

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

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20854) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 19, 1996. No significant hazards consideration comments received: No

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

Tennessee Valley Authority, Docket Nos. 50-390 Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of application for amendment: February 28, as supplemented April 15, and June 3, 1996.

Brief description of amendment: The proposed amendment would revise the Technical Specifications (TS) to increase the surveillance intervals for ice bed weight sampling and flow passage inspection from 9 months to 18 months. The TS would also be changed to provide an increased ice sublimation allowance, associated with the increased surveillance interval, by increasing the minimum total ice weight from 2,360,875 pounds to 2,403,800 pounds (1214 pounds/basket to 1236 pounds/basket).

Date of issuance: June 13, 1996

Effective date: June 13, 1996

Amendment No.: 2

Facility Operating License No. NPF-90: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: April 10, 1996 (61 FR 15998) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 13, 1996. No significant hazards consideration comments received: No

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

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, OES Nuclear, Inc., Pennsylvania Power Company, Toledo Edison Company, Docket No. 50-440 Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio

Date of application for amendment: April 26, 1996

Brief description of amendment: The amendment corrected minor technical and administrative errors in the Improved Technical Specifications prior to its implementation.

Date of issuance: June 18, 1996

Effective date: June 18, 1996

Amendment No.: 85

Facility Operating License No. NPF-58: This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: May 9, 1996 (61 FR 21213) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 18, 1996. No significant hazards consideration comments received: No

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, Ohio 44081

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

Date of amendment requests: April 25 (TXX-94119) and August 12, 1994 (TXX-94216), as supplemented by letters dated February 15 (TXX-96055), March 7 (TXX-96078), and April 11, 1996 (TXX-96111).

Brief description of amendments: These amendments modified the Administrative Controls specifications, relocate/remove requirements that are adequately controlled by existing regulations other than 10 CFR 50.36 and the technical specifications. Guidance on the proposed changes was developed by NRC and provided in the Standard Technical Specifications for Westinghouse Plants, NUREG-1431. The changes also update unit staff qualification requirements to Regulatory Guide 1.8, Revision 2.

Date of issuance: June 12, 1996

Effective date: June 12, 1996, to be implemented witnin 60 days.

Amendment Nos.: Unit 1 - Amendment No. 50; Unit 2 - Amendment No. 36

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

Date of initial notice in Federal Register: August 3, 1994 (59 FR 39599) and September 28, 1994 (59 FR 49439). The additional information contained in the supplemental letters dated February 15, March 7, and April 11, 1996, were 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 amendments is contained in a Safety Evaluation dated June 12, 1996. No significant hazards consideration comments received: No

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

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

Date of amendment request: March 12, 1996 (TXX-96008)

Brief description of amendments: The amendments revised the Technical Specifications to reflect the approval for the licensee to use of the new Containment Leakage Rate Testing Program as required by 10 CFR Part 50, Appendix J, Option B for Comanche Peak Steam Electric Station, Units 1 and 2. Implementation of the new performance based leakage rate testing program will be based on the guidance provided by Regulatory Guide 1.163, September 1995.

Date of issuance: June 13, 1996

Effective date: June 13, 1996, to be implemented within 60 days

Amendment Nos.: 51 and 37

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

Date of initial notice in Federal Register: April 10, 1996 (61 FR 15999) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 13, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendment: September 9, 1994, as superseded by letter dated July 25, 1995, and subsequently supplemented by letters dated February 28, 1996, and April 9, 1996.

Brief description of amendment: The amendment would revise TS 3/4.8.1 and its associated Bases to improve the overall emergency diesel generator reliability and availability.

Date of issuance: June 17, 1996

Effective date: June 17, 1996, to be implemented within 30 days of the date of issuance.

Amendment No.: 112

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

Date of initial notice in Federal Register: August 30, 1995 (60 FR 45188) The February 28, 1996, and April 9, 1996 supplemental letters provided additional clarifying information and did not change the staff's original no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 17, 1996. No significant hazards consideration comments received: No.

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

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: April 15, 1996

Brief description of amendments: These amendments would revise the Technical Specifications to indicate that the quadrant power tilt ratio requirements are applicable only at power levels greater than 50% of rated core power.

Date of issuance: June 7, 1996

Effective date: June 7, 1996

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

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20860) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 7, 1996. No significant hazards consideration comments received: No

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Dated at Rockville, Maryland, this 26th day of June 1996.

For the Nuclear Regulatory Commission

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