Dated: August 3, 2000.

Andrew L. Bates,

Advisory Committee Management Officer.

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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 July 15, 2000, through July 28, 2000. The last biweekly notice was published on July 26, 2000.

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 September 8, 2000, 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 electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). 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. 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 electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

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 5, 2000.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.7.8 to change the Required Actions and Completion Times for the Ultimate Heat Sink (UHS) in the event the service water (SW) temperature exceeds the 97°F surveillance acceptance limit.

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

Carolina Power & Light (CP&L) Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. The CP&L conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

The proposed change does not involve any physical alteration of plant systems, structures or components. The proposed change provides Required Actions for the plant condition where SW temperature exceeds the TS limit. The SW system temperature is not assumed to be an initiating condition of any accident analysis evaluated in the safety analysis report (SAR). Therefore, the revised limitations for SW temperature to be in excess of the design limit does not involve an increase in the probability of an accident previously evaluated in the safety analysis report. The SW system supports operability of safety-related systems used to mitigate the consequences of an accident. Plant equipment has been analyzed and determined able to perform its safety-related function at [an] SW temperature of 99°F. Performance of the containment has been analyzed in support of Amendment No. 187 to Technical Specifications assuming 100°F service water temperature and the results were acceptable. The magnitude of any increase in SW temperature in excess of the TS limit is expected to be small based on historical data and experience for the UHS. An evaluation would be performed to assure required cooling capability. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the SAR.

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 involve any physical alteration of plant systems, structures or components. The temperature of the SW when near or slightly above the design temperature does not introduce new failure mechanisms for systems, structures or components not already considered in the SAR. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

The proposed change will not allow continued operation with the SW temperature above the design basis limit. The proposed change will allow continued operation provided the required cooling capacity is verified and periodic monitoring is invoked to verify the SW temperature remains less than or equal to 99°F. Design margins are affected which are associated with systems, structures and components which are cooled by the SW system, and system temperature is an input assumption for mitigating the effects of a DBA [design-basis accident]. However, allowing SW temperature to exceed the surveillance acceptance limit, as long as required cooling is verified, will not significantly reduce the margin of safety associated with this proposed change.

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.

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: May 31, 2000.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) to delete the requirement to remove the Reactor Protection System (RPS) circuitry shorting links from TS Section 3/4.3.1, ''Reactor Protection System Instrumentation,'' 3/4.9.2, ''Refueling Operations Instrumentation,'' and 3/4.10.3, ''Shutdown Margin Demonstrations,'' and to increase the required signal-to-noise ratio for the source range monitor in (SRM) TS Sections 3/4.3.7.6, ''Source Range Monitors,'' and 3/4.9.2.

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

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

The proposed changes to TS Section 3/4.3.1, 3/4.9.2, and 3/4.10.3 will relocate the requirement that the shorting links be removed from the RPS circuitry prior to and during specified plant conditions. The removal or installation of the RPS circuitry shorting links does not have an effect on the probability of any accident previously evaluated. The proposed changes to TS Sections 3/4.3.7.6 and 3/4.9.2 will increase the minimum signal-to-noise ratio from ≥ 2:1 to ≥ 20:1, when the SRM count rate is greater than or equal to 0.7 counts per second (cps) and less than 3 cps.

The operation of the SRM does not have an effect on the probability of any accident previously evaluated. Thus, the probability of any accident previously evaluated is not increased.

The proposed changes do not affect the integrity of the fuel cladding, reactor coolant system or secondary containment, because no credit is taken in the current accident analyses for removal of the RPS circuitry shorting links. Thus, the radiological consequences of any accident previously evaluated are not increased.

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

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

The proposed changes do not affect the assumed accident performance of any LaSalle County Station structure, system or component previously evaluated because accidents previously evaluated assumed that the RPS circuitry shorting links were installed and did not credit SRM operation. The proposed changes do not introduce any new modes of system operation or failure mechanisms.

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

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

The proposed changes to TS Sections 3/4.3.1, 3/4.9.2, and 3/4.10.3 will relocate the requirement that the shorting links be removed from the RPS circuitry prior to and during specified plant conditions. The removal of the RPS circuitry shorting links in Operations Condition 5, ''Refueling,'' modifies the RPS by reconfiguring the scram signal for the intermediate range monitors (IRMs) and average power range monitors (APRMs) to non-coincidental and enabling the SRM non-coincidental high flux scram signal. However, the SRM non-coincidental high flux scram signal is not credited in any Design Basis Accident (DBA) and the IRM and APRM one-out-of-two taken twice full scram provides the credited protection with respect to safety analysis.

Refueling interlocks and shutdown margin requirements ensure that the reactor is maintained in a subcritical condition in Operational Condition 5. The refueling interlocks are required to be operable by TS Section 3/4.9.1, ''Reactor Mode Switch.'' The SRM, IRM, and APRM control rod withdrawal block interlocks are not affected by the removal or installation of the RPS circuitry shorting links. Although shutdown margin may not yet have been demonstrated in Operational Condition 5, shutdown margin calculations performed prior to altering the reactor core, along with procedural compliance for any Core Alterations, provides indication that shutdown margin is available.

The proposed changes to relocate the description and function of the RPS circuitry shorting links to the UFSAR and be controlled in accordance with the requirements of 10 CFR 50.59, are consistent with the requirements of 10 CFR 50.36, ''Technical Specifications.'' The existing TS requirements to remove the RPS circuitry shorting links do not satisfy any of the four criteria of 10 CFR 50.36 for inclusion of a requirement into the TS. In accordance with NRC guidance, existing TS requirements that do not satisfy the criteria of 10 CFR 50.36 can be removed from the TS and relocated to other controlled documents, such as the UFSAR. Changes to the LaSalle County Station UFSAR are controlled in accordance with the requirements of 10 CFR 50.59.

The proposed changes to TS Sections 3/4.3.7.6 and 3/4.9.2 will increase the statistical neutron monitoring confidence that the indicated signal is correct when the SRMs indicate in the range form 0.7 cps to 3 cps. A SRM signal-to-noise ratio of ≥ 2:1 provides a statistical neutron monitoring confidence of 95% that the indicated signal is correct with a minimum count rate of 3 cps. A study was performed which concluded that a SRM signal-to-noise ratio of ≥20:1 is required to provide a statistical neutron monitoring confidence of 95% that the indicated signal is correct at 0.7 cps.

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

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

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: December 27, 1999.

Description of amendment request: The proposed amendment would revise the technical specifications to increase the allowable out-of-service times and surveillance test intervals for selected actuation instrumentation.

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

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

The proposed TS [technical specification] changes increases the Allowable Outage Times and Surveillance Test Intervals (AOT/STI) for actuation instrumentation based on analyses developed and approved by the Nuclear Regulatory Commission (NRC). TS requirements that govern operability or routine testing of plant instruments are not assumed to be initiators of any analyzed event because these instruments are intended to prevent, detect, or mitigate accidents. Therefore, these changes will not involve an increase in the probability of occurrence of an accident previously evaluated. Additionally, these changes will not increase the consequences of an accident previously evaluated because the proposed changes do not involve any physical changes to plant systems, structures or components (SSCs), or the manner in which these SSCs are operated. These changes will not alter the operation of equipment assumed to be available for the mitigation of accidents or transients by the plant safety analysis or licensing basis. As justified and approved in the AOT/STI licensing topical reports, the proposed changes establish or maintain adequate assurance that components are operable when necessary for the prevention or mitigation of accidents or transients and that plant variables are maintained within limits necessary to satisfy the assumptions for initial conditions in the safety analyses. The proposed changes establish or modify time limits allowable for operation with inoperable instrument channels based on analyses which have been approved by the NRC. Furthermore, there will be no change in the types or significant increase in the amounts of any effluents released offsite. For these reasons, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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

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

The proposed changes increase the STIs and AOTs for actuation instrumentation based on generic analyses completed by the Boiling Water Reactor Owners' Group (BWROG). The NRC has reviewed and approved the generic studies and has concurred with the BWROG that the proposed changes do not significantly affect the probability of failure or availability of the affected instrumentation systems. The analysis determined that there is no significant change in the availability and/or reliability of instrumentation as a result of the proposed changes in STIs and AOTs. Furthermore, the change to increase the frequency of the reactor protection system scram contactor testing has been shown to improve plant safety. ComEd has determined these studies are applicable to Quad Cities Nuclear Power Station, Units 1 and 2. The proposed changes to AOTs provide realistic times to complete required testing and maintenance actions without increasing the overall instrument failure frequency. Likewise, the extended STIs do not result in significant changes in the probability of instrument failure. Furthermore, the proposed changes will reduce the probability of test-induced plant transients and equipment failures. Therefore, it is concluded that the proposed changes 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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: December 30, 1999.

Description of amendment request: The proposed amendment would revise the technical specifications to (1) remove the Main Steam Line Radiation Monitor (MSLRM) scram and main steam line isolation functions, and (2) add a new requirement for the MSLRM mechanical vacuum pump trip function.

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:

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

This proposed change involves the removal of existing Main Steam Line Radiation Monitor (MSLRM) scram and the MSLRM MSL [main steam line] Valve closure signal. The purpose of the MSLRM reactor scram and the MSL isolation signal is to mitigate the radiological effects of a fuel element failure. These functions do not serve as initiators for any of the accidents evaluated in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR). Removal of these functions will not increase the probability of any of the accidents previously evaluated.

The radiological effects of a Control Rod Drop Accident (CRDA) have been evaluated for the Boiling Water Reactor Owners' Group (BWROG) by General Electric (GE) in Report NEDO–31400A, ''Safety Evaluation For Eliminating the Boiling Water Reactor Main Steam Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor.'' The GE report was evaluated by the NRC and found acceptable by letter dated May 15, 1991, ''Acceptance for Referencing of Licensing Topical Report NEDO–31400.'' The NRC Safety Evaluation Report accepting the GE report required licensees to demonstrate that the assumptions of the GE report analysis were bounding for their plants. ComEd has evaluated the GE analysis for applicability to Quad Cities Nuclear Power Station, Units 1 and 2.

The GE analysis demonstrates that operation with the proposed change does not represent a significant increase in the consequences of a CRDA. Therefore, operation of Quad Cities Nuclear Power Station, Units 1 and 2, under the proposed change does not represent a significant increase in the probability or consequences of an accident previously evaluated. A site specific radiological evaluation was completed to confirm the applicability of the generic GE analysis to Quad Cities Nuclear Power Station.

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

This proposed change involves the removal of the existing MSLRM scram and the MSL Valve closure input from the MSL Tunnel High Radiation signal. Removal of these functions does not represent a change in operating parameters for Quad Cities Nuclear Power Station, Units 1 and 2. Removal of these functions does not add any additional hardware and does not represent any new failure modes. Operation of Quad Cities Nuclear Power Station, Units 1 and 2, under the proposed change does not create the possibility of a new or different type of accident previously evaluated.

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

The proposed change involves the elimination of the MSLRM scram and the MSL Valve closure input from the MSL Tunnel High Radiation signal. Operation under the proposed change will not change any plant operation parameters, nor any protective system setpoints other than removal of these functions. The GE report has demonstrated that the consequences of the CRDA without the MSLRM High scram and MSL Valve closure signal from the MSL Tunnel Radiation detector results in doses which are well within 10 CFR part 100, ''Reactor Site Criteria,'' limits. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: June 5, 2000.

Description of amendment request: The proposed amendment, which changes the Perry Nuclear Power Plant as described in the Updated Safety Analysis Report, modifies the circuitry to the Reactor Core Isolation Cooling (RCIC) System initiation logic. The proposed circuit modification will include a time delay to the main turbine and feedwater pump turbine trip signal associated with a RCIC system automatic initiation. The addition of this time delay will prevent potential main turbine and feedwater pump turbine trips that result in unnecessary reactor scrams from inadvertent RCIC initiations.

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 Reactor Core Isolation Cooling (RCIC) initiation turbine trip circuit performs an operational protection of the main turbine for commercial and reliability purposes. The proposed modification slightly alters the methodology by which the turbine protective features are performed but they have no influence on any of the accidents previously evaluated. The associated circuits do not interfere with higher priority protection systems.

Installation of circuits associated with the proposed modification cannot initiate an accident, nor are they used to mitigate the consequences of any previously defined accident. Their function is to provide turbine protection that is separate and distinct from the turbine overspeed protection system. The circuits modified by this modification will still result in actions taken (auto or manual) that meet the bases for the present design. Also, this modification does not alter or adversely affect the turbine overspeed function in any manner.

The proposed modification reduces the probability of occurrence of spurious turbine trips due to spurious RCIC initiation. Therefore, with the implementation of this modification, the boundaries of the accident analysis will be less challenged and result in fewer false scrams.

The proposed modification provides assurance for compliance with the current licensing basis regarding dose limits of General Design Criteria (GDC) 19 of Appendix A to 10 CFR [Part] 50 and 10 CFR [Part] 100. The proposed modification ensures originally stated design criteria are met and therefore does not affect the precursors for accidents or transients analyzed in Chapter 15 of the Perry Nuclear Power Plant (PNPP) Updated Safety Analysis Report (USAR). With the proposed modification, the radiological consequences are the same as previously stated in the USAR. Therefore, the implementation of the proposed modification does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The USAR addresses accident analysis of the reactor based on events such as turbine trips, including spurious trips and turbine missiles. The present RCIC initiation turbine trip circuit is a potential contributor to spurious turbine trips. The addition of the time delay relay reduces this potential. A time delay relay failure that fails to trip the turbine would have the same effect on the turbine as the failure of the present trip circuit that has no time delay relay. The consequence of the failure of this circuit to protect the turbine remains unchanged with the addition of a time delay relay and is bounded by the existing accident analysis. The accident analysis for missile protection of those systems, structures, components required for the safe shutdown of the plant remain unchanged.

The probability of external missile generation has not changed with implementation of the proposed modification. The Main Turbine casing and surrounding structures will not be changed by the proposed modification. The location of equipment important to safety as it relates to the turbine missiles will not be changed. Therefore the missile strike probability will not be increased by the 4½ minute time delay.

The proposed modification provides assurance for compliance with the current licensing basis regarding dose limits of GDC 19 of Appendix A to 10 CFR [Part] 50 and 10 CFR [Part] 100. The proposed modification does not change the assumptions used in any accident analysis and no new or different kind of accident is created. The proposed modification ensures originally stated design criteria are met and therefore does not affect the precursors for accidents or transients analyzed in Chapter 15 of the PNPP USAR. Therefore, the implementation of the proposed modification does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety by which this modification is evaluated against is the design/criteria of the turbine overspeed protective system relative to the PNPP USAR, SER, GDC4 and Reg[ulatory] Guide 1.115, [''Protection Against Low-Trajectory Turbine Missiles.''] The change in response time of the main turbine RCIC initiation trip circuit does not affect the margin of safety as reflected in these documents. There is no safety margin criteria associated with this circuit, as defined in the USAR or the bases for any Technical Specifications.

Although there is no margin of safety associated with the turbine, the regulatory requirement for acceptance of the turbine for use at PNPP is based upon a calculated value of probability of external turbine missile interaction with safety related equipment.

The barriers (Turbine casing and surrounding structures) and barrier interaction as previously analyzed will not be changed by this modification. The location of safety related equipment as it relates to the turbine missiles will not be changed. The probability of external missile generation has not changed with implementation of the proposed modification. Therefore, there is no reduction in the margin of safety by the proposed modification.

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.

Attorney for licensee: Mary E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: July 19, 2000.

Description of amendment request: To revise the license: (1) to implement Siemens Power Corporation (SPC) high thermal performance (HTP) fuel assembly design in Cycle 17, (2) relocate shutdown margin (SDM) requirements in Modes 1 to 5 to the Core Operating Limits Report (COLR), (3) update the COLR methodologies listed in the Technical Specification (TS) Section 6.9.1.11, and (4) request relief from the SPC fuel assembly reconstitution restrictions for peripheral low power fuel assemblies. Applicable TS surveillance requirements are changed to be consistent with the proposed license amendment. Additionally, administrative changes are proposed to the boron concentration specifications related to the boration requirements.

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

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

The proposed amendment would allow the implementation of HTP fuel design for Cycle 17. The design of this fuel will be evaluated to meet all the mechanical, neutronics and thermal-hydraulics requirements, and acceptance criteria based on the approved methodology. The relocation of shutdown margin to the COLR and other proposed changes have no adverse impact on the operation of the plant and have no relevance to the accident initiators. There are no changes to the plant configuration, and thus the frequency of occurrence of previously analyzed accidents is not affected by the proposed changes. The changes proposed to the fuel reconstitution methodology would not impact the design acceptance criteria for the reconstituted fuel assemblies.

The proposed change for the relocation of shutdown margin to the COLR has no impact on current safety analyses and their consequences. Changes to the COLR limits will be controlled per Generic Letter 88–16 under the provisions of 10 CFR 50.59 and the requirements of TS 6.9.1.11.c. The application of the added methodology, which includes the approved HTP DNB [departure from nucleate boiling] correlation, would remain consistent with the design basis requirements and would not involve a significant increase in the consequences of design basis accidents. Other proposed TS and TS bases changes do not affect safety analysis results. The changes proposed to the fuel reconstitution methodology would not impact the safety analysis consequences as the changes are related to the non-limiting rod locations.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Use of the modified specification would not create the possibility of a new or different

kind of accident from any previously evaluated.

The proposed amendment updates the list of approved methodology in TS 6.9.1.11, relocates shutdown margin requirements to the COLR and requests relief for fuel reconstitution requirements. None of these changes would create the possibility of a new kind of accident since the reload analysis with these changes would continue to meet all applicable design limits. There is no change to plant configuration, systems or components which would create new failure modes. The modes of operation of the plant would remain unchanged.

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

(3) Use of the modified specification would not involve a significant reduction in a margin of safety.

The proposed changes have no significant adverse impact on the safety analysis. As such, these changes would continue to provide margin to the acceptance criteria for specified acceptable fuel design limits (SAFDL), 10 CFR 50.46(b) requirements, primary and secondary overpressurization, peak containment pressure, potential radioactive releases, and existing limiting conditions for operation. The future use of updated approved methodologies will follow all design basis requirements to ensure that a safety margin to the acceptance criteria would continue to remain available for full power operation of St. Lucie Unit 1.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

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

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Richard P. Correia.

Florida Power and Light Company (FPL), Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of amendment request: July 19, 2000.

Description of amendment request: The amendment would revise the St. Lucie Unit 1 Technical Specifications (TS) to require laboratory testing of activated charcoal samples for applicable engineered safety feature ventilation systems using the ASTM D3803–1989 protocol. In addition the proposed changes revise the TS test criteria for methyl iodide removal efficiency to be consistent with the guidance of NRC Generic Letter (GL) 99–02. The affected Unit 1 TS are the shield building ventilation system (SBVS), TS 4.6.6.1; control room emergency ventilation system (CREVS), TS 4.7.7.1; emergency core cooling system (ECCS) area ventilation system, TS 4.7.8.1; and fuel pool ventilation system—fuel storage, TS 4.9.12.

The July 19, 2000, application is a complete replacement of the proposed Unit 1 TS amendment previously submitted by FPL letter L–99–241 on November 17, 1999. The NRC staff had previously published a **Federal Register** notice on January 12, 2000 (Vol. 65, page 1923), regarding the proposed amendments for St. Lucie Units 1 and 2, but subsequently, issued the licence amendment for St. Lucie, Unit 2 only, on February 17, 2000. This revised amendment request increases the TS-required removal efficiency of the Unit 1 SBVS, ECCS area ventilation system, and CREVS charcoal adsorbers to 97.5% when tested in accordance with ASTM D3803–1989 at 30°C, 70% relative humidity. The revised testing requirements align the TS acceptance criteria and methodology with the Unit 1 accident analysis assumptions and GL 99–02 recommendations.

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

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

The proposed amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated. The new charcoal testing protocol is performed offsite on samples extracted from the safety related ventilation systems. Therefore, there is no impact on any accident initiator and results in no changes in the probability. The proposed testing protocol is more conservative than previous tests; therefore, the efficiency of charcoal for the affected safety related systems would not be overestimated. With the new testing protocol, more conservative testing results are expected since the temperature at which testing is performed is lower and the charcoal retention capability is more consistent with actual accident conditions. The proposed change thus ensures that the charcoal in service will comply with the penetration requirements to meet the design basis accident conditions.

Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed charcoal testing protocol only affects surveillance testing requirements for safety related ventilation systems. The functions of these systems remain unchanged and unaffected. No new system interactions have been introduced by the proposed amendment, which would create a new or different type of accident than previously analyzed. No physical changes are being made to any structure, system, or component. The operation of the facility will not be altered by the proposed amendment. The systems involved are not initiators of any accidents as previously evaluated.

The proposed amendment will not change the physical plant or the modes of operation defined in the facility license. The changes do not involve the addition of new equipment or the modification of existing equipment, nor do they alter the design of St. Lucie Unit 1 systems. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendment does not involve a reduction in the margin of safety. The margin of safety of the Technical Specifications, its Bases, the Updated Final Safety Analysis Report, the Safety Evaluation Report or in any other design document has been increased by the use of a safety factor of two for the TS affected by the proposed amendment. The change provided in this proposed amendment is related to introducing an improved testing protocol for the activated charcoal in safety related ventilation systems. The change consists of testing the charcoal with a new testing protocol, higher efficiencies, and with lower test temperatures to more closely reflect accident conditions and to eliminate potential overestimation of charcoal efficiency.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

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

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: June 21, 2000.

Description of amendment request: The proposed amendments would relocate Technical Specification Surveillance Requirement (SR) 4.8.1.1.2.e.1 to a licensee controlled maintenance program that will be incorporated by reference into the next revision of each unit's Updated Final Safety Analysis Report (UFSAR). SR 4.8.1.1.2.e.1 requires that the emergency diesel generator (EDG) be inspected in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service, at least once every 18 months during shutdown. Upon relocation to the licensee controlled maintenance program the requirement to perform the EDG inspections every 18 months during shutdown will be eliminated. These amendments, in combination with the previously submitted EDG risk informed allowed outage time extension to 14 days, allows the EDG maintenance to be performed in Modes 1 and 2. The licensee stated that approval of these amendments is expected to reduce the complexity of activities performed during refueling outages and, consequently, reduce human errors.

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. There are no changes to the emergency diesel generator (EDG) maintenance program. The actual EDG maintenance program is unaffected.

The only substantive change allows the periodic EDG inspection to be performed in any operational mode instead of only during shutdown. By FPL Letter L–99–228, dated November 17, 1999, FPL has previously submitted a request for a risk informed EDG allowed outage time (AOT) extension from 3 days to 14 days. An evaluation of the impact on plant risk as expressed by the change in core damage frequency (CDF), the incremental conditional core damage probability (ICCDP), the change in large early release frequency (LERF), and the incremental conditional large early release probability (ICLERP) was provided as part of the EDG AOT extension submittal (L–99–228). The EDG downtime (hours/train/year) assumed in the EDG AOT extension risk assessment includes the out-of-service time that would be incurred due to performing the proposed EDG inspections and overhauls in Modes 1 and 2 instead of during shutdown. The risk assessment for the proposed EDG AOT extension bounds the risk for this change.

NRC Regulatory Guide (RG) 1.177, An Approach for Plant-Specific Risk-Informed Decision making: Technical Specifications, states that an ICCDP of <5.0E–07 and an ICLERP of <5.0E–08 is considered small for a single AOT change. Both the ICCDP and ICLERP for the proposed EDG AOT extension and these proposed changes are below the RG 1.177 specified values and are thus considered small.

NRC RG 1.174, An Approach for Using Probabilistic Risk Assessment in Decisions on Plant Specific Changes to the Licensing Basis, discusses acceptance criteria for changes in CDF and LERF. A change in CDF of <1E–06 with a total CDF of <1E–04/year and a change in LERF of <1E–07 with a total LERF of <1E–05 are considered very small. The changes in CDF and LERF for the EDG AOT extension and these proposed changes are below the RG 1.174 criteria and are thus considered very small.

The removal of the Mode restrictions from the maintenance program are bounded by the risk assessment for the EDG AOT extension and therefore do not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Use of the modified specification would not create the possibility of a new or different kind of accident from any previously evaluated.

The use of the modified specifications cannot 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 implementation of this administrative change 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) Use of the modified specification would not involve a significant reduction in a margin of safety.

The operating limits and functional capabilities of the affected systems, structures, and components remain unchanged by the proposed amendments. Therefore, these changes do not involve a significant reduction in the margin of safety. When the full scope of plant risk is considered, the risks incurred by performing either corrective or preventive EDG maintenance during power operation will be substantially offset by plant benefits associated with avoiding unnecessary plant transitions and/or reducing risks during shutdown operations.

Based on the above, we have determined that the proposed amendments do not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the probability of a new or different kind of accident from any previously evaluated, or (3) involve a significant reduction in a margin of safety; and therefore does 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: May 22, 2000.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) to incorporate the requirements specified in the American Society of Mechanical Engineers (ASME), Section XI, Subsection IWL, as modified and supplemented by the requirements in Section 50.55a(b)(2)(viii), Examination of concrete containments. In this regard, TS Section 3.6.1.6, "Limiting Condition for Operation," will be revised to conform to IWL tendon lift-off force requirements, and TS Sections 4.6.1.6.1, 4.6.1.6.2, and 4.6.1.6.3 will be revised to conform to containment tendon and containment surface inspection requirements specified in ASME Section XI, Subsection IWL, 1992 Edition with the 1992 Addenda, and 10 CFR 50.55a(b)(2)(viii).

The NRC Final Rule (61 FR 41303), dated August 8, 1996, requires implementation of the revised requirements for containment examination by September 9, 2001. FPL is planning to perform the containment tendon surveillance for Turkey Point Units 3 and 4 in March 2001.

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.

Approval and implementation of this amendment will have no effect on the probability or consequences of accident previously evaluated. The containment is not an accident initiating system or structure; therefore, there will be no impact on any accident probabilities by the approval of this amendment. The containment examination requirements in the proposed amendments are identical, equivalent, or more rigorous than previous requirements. The containment serves an important function to mitigate consequences of postulated accidents evaluated and the examinations proposed in this amendment will not result in a reduction in the capability of the containment to meet its intended design function. Additionally, the proposed changes to the Technical Specifications reflect the adoption of ASME Section XI Subsection IWL containment inservice inspections required by 10 CFR 55a(b)(2).

Based on the above, it is concluded that the proposed amendments do not involve a significant increase in the probability or consequences of any 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 previously evaluated.

The proposed changes do not alter the design, physical configuration, or modes of operation of the plant. No changes are being made to the plant that would introduce any new accident causal mechanisms. The proposed Technical Specification changes do not impact any plant systems that are accident initiators, since the containment functions primarily as an accident mitigator and the functional requirements of the containment structure are not changed. No new accident causal mechanisms are created as a result of NRC approval of the proposed amendments request. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation, including the performance of the containment. The containment is capable of performing as intended, and its function is verified by visual examination, post-tensioning system examinations, and leakage rate testing. The containment examination requirements in the proposed amendments are identical, equivalent, or more rigorous than previous requirements. As such, the ability of the containment to perform its design function will not be impaired by the implementation of the proposed amendments request. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

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

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: July 7, 2000.

Description of amendment request: The proposed amendments would revise the pressure-temperature (P/T) limits specified in Technical Specification (TS) 3.4.9.1 and Figures 3.4–2, 3.4–3 and 3.4–4 to extend their service period to a maximum of 32 effective full power years. Also, the proposed amendments will revise TS 3.4.9.3, Cold Overpressure Mitigation System (COMS) setpoints and its associated Surveillance Requirements 4.4.9.3.1a and 4.4.9.3.1d. COMS is the Westinghouse version of Low Temperature Overpressure Protection. Additionally, the licensee's submittal requested two exemptions from the requirements of 10 CFR 50.60 based on the American Society of Mechanical Engineers (ASME) Section XI, Code Cases N–588, ''Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1'' and N–641, ''Alternative Pressure Temperature Relationship and Low Temperature Overpressure Protection (LTOP) System Requirements, Section XI, Division 1.'' The exemption requests will be evaluated separately from the proposed license amendments.

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 probability of occurrence of an accident previously evaluated for Turkey Point is not altered by the proposed amendment to the Technical Specifications. Each accident in the Turkey Point UFSAR [Updated Final Safety Analysis Report] was examined with respect to the changes to the proposed Pressure-Temperature (P/T) limit curves and associated Cold Overpressure Mitigation System (COMS) setpoint limitations.

The proposed changes do not impact the integrity of the reactor coolant system pressure boundary (i.e., no change in operating pressure, materials, seismic loading, etc.) and therefore does not increase the potential for the occurrence of a loss of coolant accident (LOCA). The changes do not modify the reactor coolant system pressure boundary, nor make any physical changes to the facility design, material, or construction standards. The probability of any design basis accident (DBA) is not affected by this change, nor are the consequences of any DBA affected by this change. The proposed P/T limit curves and COMS setpoint limit are not considered to be an initiator or contributor to any accident currently evaluated in the Turkey Point UFSAR.

The curves and setpoint limit were generated in accordance with approved NRC and ASME methodology. Code Cases N–588 and N–641 have ASME Code Committee approval.

Delaying performance of two of the COMS surveillances (PORV [power operated relief valve] Channel Operational Test and the backup nitrogen supply verification) until 12 hours after decreasing the RCS cold leg temperature to ≤275°F during cooldown was also evaluated with respect to the plant accident analyses. The change was determined to not represent a significant increase in the probability or consequences of an accident because a) the likelihood of a low temperature overpressure event occurring concurrently with a loss of the redundant instrument air system is sufficiently small, and b) the existing procedural controls will effectively prevent challenges to the COMS.

Additionally, delaying these surveillances for 12 hours will allow the operators to focus their attention on transitioning the plant to RHR [residual heat removal] cooling. Given the timing sequence of the RHR system entry point to the COMS enable temperature, the time extension is considered to be a prudent and safety focused change to the method of performing a plant cooldown. The proposed time extension is also consistent with the operational flexibility currently provided in NUREG–1431, Standard Technical Specifications for Westinghouse Plants.

Based on the above, it is concluded that the proposed amendment does not involve a significant increase in the probability or consequences of any 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 previously evaluated.

The proposed changes do not create a new accident scenario. The requirements for the P/T limit curves and low temperature overpressure protection have been in place for some time. The fundamental approach follows approved ASME and Westinghouse topical report methodology. The proposed curves reflect the change in material properties acknowledged and managed by regulation and an upgrade in technology, which has been approved by ASME.

Delaying performance of two of the COMS surveillances (PORV Channel Operational Test and the backup nitrogen supply verification) until 12 hours after decreasing the RCS cold leg temperature to ≤275°F during cooldown was also evaluated with respect to the plant accident analyses. The change was determined to not represent a significant increase in the probability or consequences of an accident because a) the likelihood of a low temperature overpressure event occurring concurrently with a loss of the redundant instrument air system is sufficiently small, and b) the existing procedural controls will effectively prevent challenges to the COMS.

Additionally, delaying these surveillances for 12 hours is consistent with the operational flexibility currently provided in NUREG–1431, Standard Technical Specifications for Westinghouse Plants.

Since no new failure modes are associated with the proposed changes, the activity does not create the possibility of a new or different kind of accident from any 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 Technical Specifications for P/T limit curves and COMS setpoints are expiring and must be updated. The COMS setpoint is revised to incorporate additional margin in the instrument uncertainty. Conservative ASME code methods including safety factors have been used. The material properties used are from a much larger database than in past submittals. This results in many more datapoints available for the limiting weld metal than in past submittals. A new master curve of irradiated and unirradiated materials data has been developed for Turkey Point which shows that these curves and associated setpoints are conservative and represent an increase to the margin of safety. The new setpoint limit should reduce the possibility of an inadvertent PORV actuation. They should also reduce the potential for reactor coolant pump impeller cavitation or seal damage when the pumps are operated during low temperature conditions in the RCS. Changing the COMS surveillances to allow completion up to 12 hours after decreasing RCS temperature to ≤275°F during cooldown does not result in a reduction in the margin of safety. Acceptability is based on: consistency with NUREG–1431, Standard Technical Specifications Westinghouse Plants, COT [Channel Operational Test] Surveillance Requirements; the inherent reliability and redundancy of the Turkey Point Instrument Air System; and the existing procedural controls established to prevent challenges to the LTOP System. The proposed amendments will not involve a significant reduction in the margin of safety.

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

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Richard P. Correia.

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

Date of amendment requests: May 30, 2000.

Description of amendment requests: The proposed amendments would make changes to several Technical Specifications (TSs) to reflect implementation of the revised 10 CFR Part 20, ''Standards for Protection Against Radiation.'' In addition, the licensee proposed to revise TS 6.8.4.a.7 to maintain existing instantaneous dose rate limitations in the Offsite Dose Calculation Manual. Also, the licensee proposed a revision to the requirements governing the annual tabulation of radiation exposures.

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 of occurrence or consequences of an accident previously evaluated?

The proposed changes do not physically alter any plant structures, systems, or components (SSCs), and do not affect or create new accident initiators or precursors for any accident evaluated in the Updated Final Safety Analysis Report. Therefore, the probability of an accident previously evaluated is unchanged.

The proposed changes do not affect the types or amounts of radionuclides released following an accident, or the initiation and duration of their release. The changes are administrative in nature. Therefore, the consequences of an accident previously evaluated are not increased.

Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not significantly increased.

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

The proposed changes do not physically alter any SSC and do not affect or create new accident initiators or precursors. The accident analysis assumptions and results are unchanged. No new failures or interactions have been created.

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

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

10 CFR 20.1301, Appendix I to 10 CFR 50, and 40 CFR 190 establish the controls and limitations on total effective dose equivalent to individual members of the public from effluents discharged to unrestricted areas. The proposed changes maintain established limits for radioactive liquid effluents established in 10 CFR Part 20 and limits for radioactive gaseous effluents established in the ODCM. I&M continues to comply with limits specified in 10 CFR 20.1301, Appendix I to 10 CFR 50, and 40 CFR 190. Since compliance with these regulatory requirements has not been compromised, the proposed changes do not involve a significant reduction in the margin of safety.

In summary, based upon the above evaluation, I&M has concluded that the proposed amendment involves no significant hazards consideration.

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

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: June 20, 2000.

Description of amendment request: The following technical specification (TS) changes are being proposed to provide flexibility of operation. These changes include: (1) the ability to have a standby Safety Injection (SI) pump available during Reactor Coolant System (RCS) reduced inventory conditions with the RCS pressure boundary intact; (2) realigning a footnote to clarify the allowance of an inoperable SI pump to be energized for testing or filling accumulators; (3) allowance for an additional charging pump to be made capable of injection during pump-swap operations; (4) recognition that a substantial vent area exists for cold overpressure protection when the reactor vessel head is on, and the studs are fully detensioned; (5) limit maneuvering the plant beyond Hot Shutdown when one charging pump is operable; and (6) establishes a new value for the open permissive interlock associated with the Residual Heat Removal System suction isolation 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. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below:

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

The proposed changes do not affect plant systems such that their function in the control of radiological consequences is adversely affected. The proposed changes do not adversely affect accident initiators or precursors, nor alter the design assumptions, conditions, or manner in which structures, systems, and components (SSCs) perform their intended safety function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR). The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Since there are no changes to previous accident analyses, the radiological consequences associated with these analyses remain unchanged; therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not result in a change to the design basis of any plant SSC. All equipment important to safety will operate as designed. The proposed TS changes in conjunction with administrative controls will provide adequate control measures to ensure component integrity is not challenged. The proposed changes do not cause the initiation of any accident nor create any new failure mechanisms. The changes do not result in any event previously deemed incredible being made credible. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes do not adversely affect equipment design or operation and there are no changes being made to the TS-required safety limits or safety system settings that would adversely affect plant safety. The proposed TS changes in conjunction with administrative controls will provide adequate control measures to ensure component integrity is not challenged. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: June 20, 2000.

Description of amendment request: The licensee proposes revising the Technical Specifications (TS) by removing the prescriptive requirement for determining the reactor coolant system flow rate by precision heat balance in Surveillance Requirement 4.2.5.3 and incorporating a time limit for completion of the surveillance requirement. The change would also revise TS Table 2.2–1 to reflect the allowed calibration tolerance of the protection racks and note that the Trip Setpoint for Functional Unit 12, Reactor Coolant Flow-Low reactor trip is based on an indicated value rather than a measured value.

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

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

The proposed changes do not adversely affect accident initiators or precursors nor alter the design, conditions, and configuration of the facility or the manner in which the plant is operated. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR).

Determination of RCS [Reactor Coolant System] total flow rate by elbow tap ΔP measurement will not subject the reactor core to conditions adverse to nuclear safety. The proposed change does not affect the source term; containment isolation or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated in the Seabrook Station UFSAR. The initial conditions for all accident scenarios modeled are the same. Therefore, the consequences of an accident occurring remain unchanged.

The evaluation for use of elbow tap ΔP measurement determined that sufficient margin exists to account for all reasonable instrument uncertainties, therefore no changes to installed equipment or hardware in the plant are required. Though the calibration process of the elbow tap ΔP transmitters has changed, *i.e.*, normalization to previously performed precision RCS flow calorimetrics for Cycles 1 and 2 instead of normalization to a precision RCS flow calorimetric each cycle, this has been accounted for by the addition of instrument uncertainties usually considered to be zeroed out by normalization performed each cycle. Accounting for the additional instrument uncertainties yields a flow uncertainty that is slightly less (2.3 percent) than the current NRC [Nuclear Regulatory Commission] licensed value (2.4 percent), thus no change is required to the nominal reactor trip setpoint for RCS flow. The proposed change has no adverse affect on component or system interactions. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not alter the design, conditions and configuration of the facility or the manner in which the plant is operated and maintained in a state of readiness. Existing system and component redundancy is not being changed by the proposed changes. Though the calibration process of the elbow tap ΔP transmitters has changed, i.e., normalization to previously performed precision RCS flow calorimetrics for Cycles 1 and 2 instead of normalization to a precision RCS flow calorimetric each cycle, this has been accounted for by the addition of instrument uncertainties usually considered to be zeroed out by normalization performed each cycle. Accounting for the additional instrument uncertainties yields a flow uncertainty that is slightly less than the current NRC licensed value, thus no change is required to the nominal reactor trip setpoint for RCS flow. The proposed change has no adverse affect on component or system interactions. The time of reactor trip remains the same. Therefore, since there are no changes to the design, conditions, configuration of the facility, or the manner in which the plant is operated and maintained in a state of readiness, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed changes do not adversely affect equipment design or operation and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. The additional instrument uncertainties resulting from use of elbow tap ΔP transmitters without the requirement to normalize to a precision RCS flow calorimetric each cycle have been accounted for and no change in the nominal Trip Setpoint is required. The calculated instrument uncertainty is 2.3 percent flow. This uncertainty is slightly less than the current licensed value of 2.4 percent flow. The time of reactor trip, as modeled in the various safety analyses, is maintained. Therefore, there is no significant reduction in a margin of safety.

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Section Chief: James W. Clifford.

Northeast Nuclear Energy Company, et al., Docket Nos. 50–336 and 50–423, Millstone Nuclear Power Station, Unit Nos. 2 and 3, New London County, Connecticut

Date of amendment request: February 22, 2000

Description of amendment request: The proposed changes to the Technical Specifications (TSs) are associated with radiological effluent. The proposed changes will relocate selected radiological effluent TSs and the associated Bases to the Millstone Radiological Effluent Monitoring and Offsite Dose Calculation Manual in accordance with the Nuclear Regulatory Commission's (NRC) Generic Letter 89–01.

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

In accordance with 10 CFR 50.92, NNECO [Northeast Nuclear Energy Company] has reviewed the proposed changes and has concluded that they do not involve a Significant Hazards Consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

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

The purpose of the Radiological Liquid and Gaseous Effluent Monitoring Instrumentation is to monitor routine radioactive releases. [This] instrumentation provide[s] a surveillance of potential release points and initiates automatic alarm and trip functions which will terminate the release prior to exceeding the limits of 10 CFR Part 20 (1993 version). Relocation of Technical Specification 3.3.3.9, ''Radioactive Liquid Effluent Monitoring Instrumentation,'' and Technical Specification 3.3.3.10, ''Radioactive Gaseous Effluent Monitoring Instrumentation,'' to the Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM) does not imply any reduction in its importance in monitoring routine radioactive releases. These instruments are neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary before a design basis accident, nor do they function as a primary success path to mitigate events which assume a failure of or a challenge to the integrity of fission product barriers. These monitors are not an active design feature needed to preclude analyzed accidents or transients. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

Technical Specification 3.11.1.1 ensure[s] the concentration of radioactive materials released in liquid waste effluents from the site will be less than the concentration levels specified in 10 CFR Part 20 (1993 version), Appendix B, Table II. Technical Specification 3.11.1.2 ensures the dose or dose commitment from radioactive materials released in liquid waste effluents will not exceed the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. Technical Specification 3.11.2.1 ensures the dose rate from gaseous effluents released from all units on site will be less than dose limits specified in 10 CFR Part 20 (1993 version), Appendix B, Table II. Technical Specification 3.11.2.2 ensures the dose from noble gases released in gaseous effluents will not exceed the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. Technical Specification 3.11.2.3 implements the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. Technical Specification 3.11.3 ensures the reporting requirements of 40 CFR 190 are met. Relocation of these Technical Specifications to REMODCM does not imply any reduction in its importance in ensuring that the regulatory limits are met. The instrumentation covered by these Technical Specifications [is] neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary before a design basis accident, nor [does it] function as a primary success path to mitigate events which assume a failure of or a challenge to the integrity of fission product barriers. [This] instrumentation [is] not an active design feature needed to preclude analyzed accidents or transients. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

As a result of the relocation of the Radiological Effluent Technical Specifications (RETS) to the REMODCM, there are no Technical Specifications remaining that use definitions 1.31 and 1.26, ''Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM),'' of Unit Nos. 2 and 3 respectively. The guidelines and procedures addressing the use of radioactive waste treatment systems are covered by Specifications 6.15 and 6.13 of unit Nos. 2 and 3 respectively, which describes the REMODCM. Therefore, definitions 1.33 and 1.25, ''Radioactive Waste Treatment Systems,'' of Unit Nos. 2 and 3 respectively are no longer needed. In addition, there are no Specifications that use this phrase in the context of a defined term. These changes do not impact the assumptions used in any accident analysis, affect plant equipment, plant configuration, or the way the plant is operated. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

Replacing Technical Specification 6.9.1.6 of Millstone Unit No. 2 with Technical Specifications 6.9.1.6a and 6.9.1.6b and revising Technical Specifications 6.9.1.3 and 6.9.1.4 of Millstone Unit No. 3 will provide descriptions which satisfy the requirements of parts 10 CFR 50.36a and 10 CFR 50, Appendix I, Sections IV.B.1, IV.B.2, IV.B.3, and IV.C. These changes are consistent with NUREG–1432 and NUREG–1431. These changes do not impact the assumptions used in any accident analysis, affect plant equipment, plant configuration, or the way the plant is operated. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

The description of the REMODCM contained in Technical Specifications 6.15 and 6.13 of Millstone Unit Nos. 2 and 3 respectively will be modified to be consistent with the guidance of GL 89–01, and with NUREG–1432 and NUREG–1431. Additional minor changes have been made to be consistent with the proposed changes to Technical Specification 6.9.1.6 of Millstone Unit No. 2 and Technical Specifications 6.9.1.3 and 6.9.1.4 of Millstone Unit No. 3. These changes do not impact the assumptions used in any accident analysis, affect plant equipment, plant configuration, or the way the plant is operated. Therefore, this change will not significantly increase the probability consequences of an accident previously evaluated.

Adding Technical Specifications 6.20 and 6.15, Radiological Effluent Controls Program, to Millstone Unit Nos. 2 and 3 respectively, and 6.21 and 6.16, Radiological Environmental Monitoring Program, to Millstone Unit Nos. 2 and 3 respectively is consistent with the guidance contained in Generic Letter 89–01 for the relocation of the Radiological Effluents Technical Specifications and with NUREG–1432 and NUREG–1431. Additional minor changes have been made to be consistent with the version of 10 CFR 20, Appendix B, Table II, Column 1 which is being used by Millstone Unit Nos. 2 and 3, namely the 1993 version. These changes do not impact the assumptions used in any accident analysis, affect plant equipment, plant configuration, or the way the plant is operated. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

The following proposed changes are administrative in nature. Therefore, these changes will not significantly increase the probability or consequences of an accident previously evaluated.

- Revise Index Pages of Unit Nos. 2 and 3 Technical Specifications to reflect the proposed changes to relocate the RETS to the REMODCM.
- Address additional changes to the Millstone Unit No. 2 Technical Specifications to resolve issues not related to transferring the RETS to the REMODCM.
- Relocate to the associated Bases sections.

The proposed changes do not alter how any structure, system, or component functions. There will be no effect on equipment important to safety. The proposed changes have no effect on any of the design basis accidents previously evaluated. Therefore, this License Amendment Request does not impact the probability of an accident previously evaluated, nor does it involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes do not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Relocation of Technical Specifications 3.3.3.9, 3.3.3.10, 3.11.1.1, 3.11.1.2, 3.11.2.1, 3.11.2.2, 3.11.2.3, and 3.11.3 to REMODCM does not imply any reduction in its importance in monitoring and ensuring that the regulatory limits are met. As a result of the relocation of the RETS to the REMODCM, there are no Technical Specifications remaining that use definitions 1.31 and 1.26. Additionally, the guidelines and procedures addressing the use of radioactive waste treatment systems which are covered by Specifications 6.15 and 6.13 remove the need for definitions 1.33 and 1.25 of Unit Nos. 2 and 3 respectively. Replacing Technical Specification 6.9.1.6 of Millstone Unit No. 2 with Technical Specifications 6.9.1.6a and 6.9.1.6b and revising Technical Specifications 6.9.1.3 and 6.9.1.4 of Millstone Unit No. 3 will provide descriptions which satisfy the requirements of parts 10 CFR 50.36a and 10 CFR 50, Appendix I, Sections IV.B.1, IV.B.2, IV.B.3, and IV.C. Modifying the description of the REMODCM contained in Technical Specifications 6.15 and 6.13 of Millstone Unit Nos. 2 and 3 respectively and adding Technical Specifications 6.20, 6.21 and 6.15, 6.16 to Millstone Unit Nos. 2 and 3 respectively is consistent with the guidance contained in Generic Letter 89–01 for the relocation of the Radiological Effluents Technical Specifications and with NUREG–1432 and NUREG–1431.

The proposed changes do not affect any of the assumptions used in the accident analysis, nor do they affect any operability requirements for equipment important to plant safety. Therefore, the proposed changes will not result in a significant reduction in the margin of safety as defined in the Bases for Technical Specifications covered in this License Amendment Request.

As described above, this License Amendment Request does not involve a significant increase in the probability of an accident previously evaluated, does not involve a significant increase in the consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident from any accident previously evaluated, and does not result in a significant reduction in a margin of safety. Therefore, NNECO has concluded that the proposed changes do not involve an SHC.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: June 26, 2000.

Description of amendment request: The proposed changes to Millstone, Unit 3, Technical Specifications (TS) revise TS Section 1.13, Definitions, ''Engineered Safety Features Response Time'', TS Section 1.28, ''Reactor Trip System Response Time,'' TS Section 3.3.1, ''Instrumentation—Reactor Trip System Instrumentation,'' and TS Section 3.3.2, ''Instrumentation—Engineered Safety Features Actuation System Instrumentation'' to provide for verification of response time for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the Nuclear Regulatory Commission.

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

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

This change to the Technical Specifications does not result in a condition where the design, material, and construction standards that were applicable prior to the change are altered. The same RTS [Reactor Trip System] and ESFAS [Emergency Safety Features Actuation System] instrumentation is being used; the time response allocations/modeling assumptions in the Chapter 15 analyses are still the same; only the method of verifying time response is changed. The proposed change will not modify any system interface and could not increase the likelihood of an accident since these events are independent of this change. The proposed activity will not change, degrade or prevent actions or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the SAR [Safety Evaluation Report]. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

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

This change does not alter the performance of the pressure and differential pressure transmitters, Process Protection racks, Nuclear Instrumentation, and Logic Systems used in the plant protection systems. These sensors and systems will still have response time verified by test before being placed in operational service. Changing the method of periodically verifying instrument response for these sensors and systems (assuring equipment operability) from time response testing or calibration and channel checks will not create any new accident initiators or scenarios. Periodic surveillance of these sensors and systems will continue and may be used to (a) detect significant degradation in the sensor responses characteristic, and (b) other degradation that could cause the response time characteristic to exceed the total allowance. The total time response allowance for each function bounds all degradation that cannot be detected by periodic surveillance. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change does not affect the total system response time assumed in the safety analysis. The periodic system response time verification method for selected pressure and differential pressure sensors, the Process Protection racks, Nuclear Instrumentation, and Logic Systems is modified to allow use of actual test data or engineering data. The method of verification still provides assurance that the total system response is within that defined in the safety analysis, since calibration tests will continue to be performed and may be used to detect any degradation which (a) might significantly affect sensor response time, or (b) might cause the response time to exceed the total allowance. The total system time response allowance for each function bounds all degradation that cannot be detected by periodic surveillance. Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to plant 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.

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

NRC Section Chief: James W. Clifford.

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: May 31, 2000.

Description of amendment request: The proposed amendments would revise the Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, Technical Specifications (TSs) Surveillance Requirement (SR) 3.6.1.3.11 to allow a representative sample of reactor instrumentation line excess flow check valves (EFCVs) to be tested every 24 months, instead of testing each EFCV every 24 months.

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

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

The current SR frequency requires each reactor instrumentation line EFCV to be tested every 24 months. The EFCVs at PBAPS, Units 2 and 3 are designed to not close accidentally during normal operation, but will close automatically in the event of a line break downstream of the valve. The proposed changes would allow a reduced number of EFCVs to be tested each operating cycle. Since the EFCVs are an accident mitigation feature, their postulated failure to isolate cannot initiate previously evaluated accidents. In addition, since the proposed changes will only change the surveillance frequency, there can be no increase in the probability of occurrence of an accident as a result of this proposed change.

The postulated break of an instrument line attached to the reactor coolant pressure boundary is discussed and evaluated in the Updated Final Safety Analysis Report (UFSAR), Section 5.2.3.5. The proposed change will continue to verify the operability of the EFCVs to perform their mitigating functions. Industry operating experience as documented in the Boiling Water Reactors Owners Group (BWROG) Report B21–00658–01 provides supporting evidence that the reduced testing frequency will not affect the high reliability of these valves. The radiation dose consequences of such a break are not impacted by this proposed change. Therefore, the proposed TS changes do not involve a significant increase in the 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 proposed changes would allow a reduced number of EFCVs to be tested each operating cycle. No other changes in requirements are being proposed. The changes are not a physical alteration of the plant and will not alter the operation of the structures, systems and components as described in the UFSAR. Therefore, a new or different kind of accident will not be created.

3. The proposed TS changes do not involve a significant reduction in a margin of safety. The consequences of an unisolable rupture of an instrument line has been previously evaluated in the PBAPS, Units 2 and 3 UFSAR, Section 5.2.3.5. That evaluation assumed a continuous discharge of reactor water for the duration of the detection and cooldown sequence. The integrity and functional performance of the secondary containment and standby gas treatment system are not impaired by this event, and the calculated potential offsite exposures are substantially below the guidelines of 10 CFR Part 100. Therefore, a failure of an EFCV, though not expected as a result of this TS change, is bounded by the previous evaluation of an instrument line break. Since the proposed changes are only affecting the surveillance frequency, the accident analyses are unaffected and this change does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: June 7, 2000.

Description of amendment request: The proposed amendment to the Indian Point Nuclear Generating Unit No. 3 (IP3) Technical Specifications (TSs) would require either the Operations Manager or the Assistant Operations Manager to hold a Senior Reactor Operator (SRO) license. The proposed amendment would also remove the title of ''Shift Manager'' from 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:

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

No. This change allows either the Operations Manager or Assistant Operations Manager to be SRO licensed. This is an administrative change. The Operations department will still have an SRO licensed individual overseeing the operating crews. Therefore, there will be no increase in the probability or consequences of an evaluated accident. This is consistent with the qualifications required to be a manager in TS 6.3.1.

The change also deletes the title of Shift Manager. At IP3, ''Shift Manager'' is the NYPA [New York Power Authority] specific title for the person meeting the requirements of 10 CFR 50.54(m)(2)(ii) as the SRO assigned responsibility for overall plant operation. This requirement is redundant to 10 CFR 50.54(m)(2)(ii) and TS section 6.2.2 requirements for an SRO and therefore removal is an administrative change with no increase in the probability or consequences of an accident.

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

No. The change allows either the Operations Manager or Assistant Operations Manager to hold the SRO license. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated since they do not affect plant configuration or plant design. The Operations Manager and the Assistant Operations Manager are still required to maintain a knowledge of IP3 plant design and operations through job position requirements.

The change also deletes the title of Shift Manager. At IP3, ''Shift Manager'' is the NYPA specific title for the person meeting the requirements of 10 CFR 50.54(m)(2)(ii) as the SRO assigned responsibility for overall plant operation. This requirement is redundant to 10 CFR 50.54(m)(2)(ii) and TS section 6.2.2 requirements and it is therefore an administrative change that cannot create the possibility of a new or different accident.

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

No. The change allows either the Operations Manager or Assistant Operations Manager to hold the SRO License. The proposed amendment does not involve a significant reduction in a margin of safety because the Operations Manager and/or the Assistant Operations Manager is still required to maintain a current SRO license. Administrative Controls ensure that shift activities are directed by an individual holding an SRO license. Technical Specification 6.3.1 ensure that the Operations Manager will be a knowledgeable and qualified individual.

The change also deletes the title of Shift Manager. At IP3, ''Shift Manager'' is the NYPA specific title for the person meeting the requirements of 10 CFR 50.54(m)(2)(ii) as the SRO assigned responsibility for overall plant operation. This requirement is redundant to 10 CFR 50.54(m)(2)(ii) therefore the change has no effect on requirements and cannot offset 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.

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

NRC Section Chief: Marsha Gamberoni.

Rochester Gas and Electric Corporation, Docket No. 50–244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: July 21, 2000.

Description of amendment request: The proposed amendment would delete the requirement to have the Control Room Emergency Air Treatment System (CREATS) Actuation Instrumentation and CREATS operable in Modes 5 and 6 except during core alterations and fuel movement.

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:

Evaluation of More Restrictive Changes

The more restrictive changes (which is a conservative characterization, as these changes are implied by the current specifications) associated with amending the Applicability section for LCO [limiting condition for operation] 3.3.6 and LCO 3.7.9, and Condition C of LCO 3.3.6 and Condition D and F of LCO 3.7.9, to include ''during CORE ALTERATIONS'', do not involve a significant hazards consideration as discussed below:

(1) Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The changes add a conservative Mode of Applicability for the Control Room Emergency Air Treatment System (CREATS) and CREATS actuation instrumentation. This does not increase the probability of an accident previously evaluated since the CREATS and CREATS actuation instrumentation themselves are not accident initiators. The proposed changes are consistent with the guidance of NUREG–1431 and provide assurance that the CREATS is in the conservative mode of operation for a response to an accident. Therefore, the probability or consequences of an accident previously evaluated are not significantly increased.

(2) Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change for a new mode of applicability does not of itself involve a physical alteration of the plant or change in the methods governing normal plant operation. The change only involves a conservative increase in the requirement of when the CREATS and CREATS actuation instrumentation are operable. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated are not created.

(3) Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed change requires the CREATS and CREATS actuation instrumentation to be in the conservative mode of operation for a response to an accident. The change adds conservatism as determined by the guidance of NUREG–1431. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the preceding information, it has been determined that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

Evaluation of Less Restrictive Changes

The less restrictive changes associated with amending the applicability sections for LCO 3.3.6 and LCO 3.7.9, and Condition C of LCO 3.3.6 and Condition D and F of LCO 3.7.9, to delete Modes 5 and 6 from these sections do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The changes are the result of an analysis performed of the control room dose consequences which could occur as the result of a potential waste gas decay tank failure. This does not increase the probability of an accident previously evaluated since the Control Room Emergency Air Treatment System (CREATS) and CREATS actuation instrumentation themselves are not accident initiators. The results of the analysis show that if no credit is taken for the CREATS, the control room doses remain well within the limits specified in 10 CFR 50, Appendix A, GDC [General Design Criteria] 19 and the guidance provided by the NRC in NUREG–0737 Section ll.B.2, Dose Rate Criteria, and NUREG–0800 Section 6.4, Control Room Habitability Program. The proposed Mode of Applicability change is consistent with the guidance of NUREG–1431 which allows plant-specific changes with respect to Modes 5 and 6. Therefore, the probability or consequences of an accident previously evaluated are not significantly increased.

(2) Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes associated with the modes of applicability for the CREATS and CREATS actuation instrumentation are not of themselves nor do they affect potential accident initiators. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated are not created.

(3) Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes remove the requirements for the control room ventilation system, which has been shown by analysis to not be required to meet regulatory limits. The changes are consistent with the guidance of NUREG–1431. Therefore, these changes do not involve a significant reduction in a margin of safety.

Based upon the preceding information, it has been determined that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

The less restrictive change associated with amending the Required Action and Completion Time of Condition C of LCO 3.3.6 and Condition F of LCO 3.7.9 to remove a required action, do not involve a significant hazards consideration as discussed below:

(1) Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes to remove a required action of restoring equipment to operable status do not affect the probability of an accident as the Control Room Emergency Air Treatment System (CREATS) and CREATS actuation instrumentation, in and of themselves, have no failure modes or effects which are precursors to accidents. The proposed changes do not introduce any new failure modes or effects to any other system or component which is a precursor to an accident. The remaining Required Actions within the referenced Conditions place the plant outside of the Mode of Applicability for these systems. Therefore, the probability or consequences of an accident previously evaluated are not significantly increased.

(2) Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes do not of themselves involve a physical alteration of the plant or change in the methods governing normal plant operation. The proposed changes create no new functional interactions with existing plant equipment nor do they introduce any new failure modes or mechanisms which could lead to reactor core damage or fission product release. Therefore, because the changes do not affect any system that can act as an accident precursor, the possibility for a new or different kind of accident from any accident previously evaluated are not created.

(3) Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes remove requirements for restoring systems which are no longer required. The changes are consistent with the guidance of NUREG–1431. Therefore, these changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Nicholas S. Reynolds, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Marsha K. Gamberoni.

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

Date of amendment request: June 29, 2000.

Description of amendment request: The proposed amendment would change the Farley Nuclear Plant, Units 1 and 2, design bases described in the Final Safety Analysis Report. The change adds a description of the methodology Southern Nuclear Operating Company uses to determine what systems and components need to be protected from tornado missiles.

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.

Proposed for NRC review and approval are changes to the Farley Nuclear Plant (FNP) Final Safety Analysis Report (FSAR) which in essence constitute a license amendment to incorporate use of an NRC[-]approved methodology to assess the need for additional positive (physical) tornado missile protection of specific features at FNP. The FSAR changes will reflect use of the Electric Power Research Institute (EPRI) Topical Report ''Tornado Missile Risk Evaluation Methodology'' (EPRI NP–2005), Volumes I and II. As noted in the NRC Safety Evaluation Report on this topic dated October 26, 1983, the current licensing criteria governing tornado missile protection are contained in Standard Review Plan (SRP) Sections 3.5.1.4 and 3.5.2. These criteria generally specify that safety-related systems be provided positive tornado missile protection (barriers) from the maximum credible tornado threat. However, SRP Section 3.5.1.4 includes acceptance criteria permitting relaxation of the above deterministic guidance, if it can be demonstrated that the probability of damage to unprotected essential safety-related features is sufficiently small.

As permitted in NRC Standard Review Plan (NUREG–0800) sections, the combined probability will be maintained below an allowable level, i.e., an acceptance criterion threshold, which reflects an extremely low probability of occurrence. The FNP approach assumes that if the probability calculation result for the total plant identifies that the probability of a combination of tornado missiles striking and damaging a portion of an important system or component is greater than or equal to 10^{-6} then installation of unique missile barriers would be needed to lower the total combined probability below the acceptance criterion of 10^{-6}.

With respect to the probability of occurrence or the consequences of an accident previously evaluated in the FSAR, the possibility of a tornado reaching the FNP site and causing damage to plant structures, systems and components is a design basis event considered in the [FSAR]. The changes being proposed do not affect the probability that the natural phenomenon (a tornado) will reach the plant, but from a licensing basis perspective they do affect the probability that missiles generated by the winds of the tornado might strike and damage certain plant systems or components. There are a limited number of safety-related components that could theoretically be struck and consequently damaged by tornado-generated missiles. The probability of tornado-generated missile strikes on ''important'' systems and components (as discussed in Regulatory Guide 1.117) is what is to be analyzed using the probability methods discussed above. The combined probability of damage will be maintained below an extremely low acceptance criterion to ensure overall plant safety. The proposed change is not considered to constitute a significant increase in the probability of occurrence or the consequences of an accident, due to the extremely low probability of damage due to tornado-generated missiles and thus an extremely low probability of a radiological release. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of previously evaluated accidents.

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

The possibility of a tornado reaching the FNP site is a design basis event considered in the [FSAR]. This change involves recognition of the acceptability of performing tornado missile probability calculations in accordance with established regulatory guidance. The change therefore deals with an established design basis event (the tornado). Therefore, the proposed change would not contribute to the possibility of a new or different kind of accident from those previously analyzed. The probability and consequences of such a design basis event are addressed in Question 1 above. Based on the above discussions, the proposed change will not create the possibility of a new or different kind of accident than those previously evaluated.

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

The existing licensing basis for FNP with respect to the design basis event of a tornado reaching the plant, generating missiles and directing them toward safety-related systems and components is to provide positive missile barriers for all safety-related systems and components. With the change, it will be recognized that there is an extremely low probability, below an established acceptance limit, that a limited subset of the ''important'' systems and components could be struck and consequently damaged. The change from protecting all safety-related systems and components to ensuring an extremely low probability of occurrence of tornado-generated missile strikes and consequential damage on portions of important systems and components is not considered to constitute a significant decrease in the margin of safety due to that extremely low probability. Therefore, the changes associated with this license amendment request do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the 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.

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama.

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendment request: May 16, 2000.

Description of amendments request: Amend Technical Specification (TS) 4.8.1.1.2 to revise the emergency diesel generator fuel oil surveillance requirements to adopt more current industry standards.

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

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

The probability of occurrence or the consequences for an accident is not increased by this request. The proposal to establish a Diesel Fuel Oil Program and specifying the ASTM [American Society for Testing and Materials] standards in the TS Bases does not modify the manner in which the plant is operated. Deletion of the portion of the surveillance requirement (SR) that specifies the use of sodium hypochlorite solution in cleaning of the fuel oil storage tanks, and the deletion of the SR to perform a pressure test of those portions of the diesel fuel oil system designed as Section III, subsection ND of the ASME [American Society of Mechanical Engineers] Code do not alter the way any structure, system, or component functions and does not modify the manner in which the plant is operated.

This request will ensure that the fuel oil continues to be properly evaluated to ensure that the fuel oil will not degrade the ability of the D/G [diesel generator] to perform its intended function. The fuel oil storage tanks will be cleaned at the required frequency. The deletion of the SR to perform a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code, removes potential confusion about testing of the fuel oil system since no portion of the system is designed to Section III, subsection ND of the ASME Code. Therefore, these changes will not change or impact previously evaluated accidents and the D/Gs ability to perform their intended function.

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

The proposed changes are procedural in nature concerning fuel oil testing, cleaning chemical to be used on the fuel oil storage tanks, and deletion of the pressure test of those portions of the diesel fuel oil system designed as Section III, subsection ND of the ASME Code. The possibility for an accident or malfunction of a different type than any evaluated previously in SQN's [Sequoyah's] Final Safety Analysis Report are not created. The proposal does not alter the way any structure, system, or component functions and does not modify the manner in which the plant is operated. The fuel oil quality will not be reduced and will not result in a decrease in D/G operability. The fuel oil storage tanks will be cleaned at the required frequency. Therefore, the possibility of a new or different kind of accident previously evaluated is not created.

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

The proposed changes are procedural in nature concerning fuel oil testing, cleaning chemical to be used on the fuel oil storage tanks, and deletion of the pressure test of the diesel fuel oil system. The margin of safety has not been reduced since the change in test methodologies are NRC approved and will continue to ensure the quality of the fuel oil. Also, deletion of the portion of the SR that specifies the use of sodium hypochlorite does not change the requirement to clean the fuel oil storage tanks. ASME Code requirements will continue to be met. Therefore, the proposed changes do not involve a reduction in the margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

Tennessee Valley Authority, Docket No. 50–390 Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of amendment request: July 10, 2000 (TS 00–08).

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) as follows:

Part A—Channel Operational Test (COT) 12 Hour Limit

Channel operational tests (COTs) are performed for the Power Range and Intermediate Range neutron monitors in accordance with Reactor Trip System (RTS) Surveillance Requirements (SRs) 3.3.1.7 and 3.3.1.8. While the unit is in Modes 1 or 2, SR 3.3.1.7 is performed for the Power Range monitors every 92 days. SR 3.3.1.8 is performed for the Intermediate Range monitors prior to startup of the reactor and at various points during power escalation or reduction. In addition, SR 3.1.10.1 currently requires that a COT be performed on the Power Range and Intermediate Range neutron monitors within 12 hours prior to initiation of a physics test, even though SR 3.3.1.7 and SR 3.3.1.8 have been performed on the required frequency.

TVA proposes to eliminate the 12 hour requirement for the testing required by SR 3.1.10.1 so that the testing performed for SR 3.3.1.7 and SR 3.3.1.8 can be used to satisfy SR 3.1.10.1. This issue was addressed by Technical Specification Task Force (TSTF) Traveler 108. The proposed amendment revises SR 3.1.10.1 to implement the portion of the approved TSTF 108 applicable to Watts Bar.

Part B—Trip System Logic for Physics Testing TSTF Traveler 315

During the performance of physics testing one power range channel is used to provide input to the reactivity computer. In preparation for the test, the fuses to the electronics drawer for the channel are removed and the channel is placed in a tripped condition and results in the NIS trip logic being in a one-out-of-three logic status. Therefore, any spurious signals received on one channel will result in a reactor trip. The changes proposed by TSTF–315 allows the fuses to remain in the NIS channel that is connected to the reactivity computer and avoid tripping the bistables associated with the NIS channel. This configuration results in the channel being in a bypassed state and places the overall logic in a two-out-of-three logic status. The advantage of this configuration is that a single spurious signal would not result in a reactor trip. The proposed amendment does not deviate from the version of TSTF–315 that was approved by NRC on June 29, 1999.

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

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

Part A—Channel Operational Test (COT) 12 Hour Limit

The proposed amendment removes the requirement to perform an additional Channel Operational Test (COT) on the Intermediate and Power Range functions within 12 hours of performing a physics test. The Intermediate and Power Range instrumentation is determined to be OPERABLE by periodic surveillance requirements which must be confirmed to be within frequency prior to making the reactor critical. A COT for the Intermediate or Power Range instrumentation is not a precursor to, or assumed to be an initiator of any analyzed accident. Therefore, this change does not involve a significant increase in the probability of an accident previously evaluated.

Regarding a significant increase in the consequences of an accident, several factors must be considered. First the physics tests are performed in accordance with the Technical Specifications in Mode 2. Therefore, the power level of the reactor is limited to 5 percent or less. Along with this, the reactor trip function of the intermediate range detectors will be unaffected by the proposed amendment and therefore, will be available to mitigate a reactivity transient at low power. Further, the trip setpoint for the power range monitors are decreased during startup of the reactor from the normal 109% setpoint to a value less than or equal to 85%. This setpoint reduction provides an additional measure to limit a reactivity excursion. Considering these factors, the proposed change will not involve a significant increase in the consequences of an accident previously evaluated.

Part B—Trip System Logic for Physics Testing

During the performance of physics testing one power range channel is used to provide input to the reactivity computer. In preparation for the test, the fuses to the electronics drawer for the channel are removed and the channel is placed in a tripped condition and results in the NIS trip logic being in a one-out-of-three logic status. Therefore, any spurious signals received on one channel will result in a reactor trip. The changes proposed by TSTF–315 allows the fuses to remain in the NIS channel that is connected to the reactivity computer. This configuration results in the channel being in a bypassed state and places the overall logic in a two-out-of-three logic status. The advantage of this configuration is that a single spurious signal will not result in a reactor trip. In addition, the physics tests required by LCO 3.1.10 are performed while the reactor is in Mode 2. Therefore, the thermal power of the reactor is restricted to 5 percent or less. Neutron flux, which is monitored by the NIS, is only one of several RTS variables which may initiate a reactor trip in Mode 2. The other variables include reactor coolant temperature, pressurizer pressure and steam generator water level. These variables are unaffected by the proposed amendment. Considering this, the low thermal power level of the reactor, and a potential reduction in unnecessary plant transients due to the one-out-of-three logic, the proposed amendment will not significantly impact the safe operation of the plant. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Part A—Channel Operational Test (COT) 12 Hour Limit

The proposed amendment is not based on a change in the design or configuration of the plant. Also, the proposed amendment does not change the manner in which the plant is operated. The amendment deletes the requirement for the performance of a COT for the Intermediate and Power Range instrumentation within 12 hours of starting a physics test. Therefore, the proposed change will not create the possibility of a new or different kind of accident than any previously evaluated.

Part B—Trip System Logic for Physics Testing

The NIS provides indication, alarm, control, and trip signals along with the capability to monitor neutron flux over the complete range from reactor shutdown to 120 percent full power. The system also generates permissive and level trip signals, which are then coupled to the logic matrices of the RTS. This interface either allows power changes based upon proper functioning of the next range of measurement instrumentation or shuts down the reactor as unsafe operating limits are approached. The changes in the operation of the NIS proposed by this amendment for TSTF–315, do not inhibit the capabilities of the system to initiate a reactor trip, if required. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident.

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

Part A—Channel Operational Test (COT) 12 Hour Limit

As stated previously, the proposed change deletes the requirement to perform an additional COT for the Intermediate and Power Range functions within 12 hours of the start of physics test. The Intermediate and Power Range instrumentation channels are determined to be operable by meeting the requirements of the periodic surveillances. These surveillance requirements are not affected by the proposed amendment. Since the equipment will be determined to be operable by periodic surveillances, the performance of the a surveillance prior to the initiation of a physics test does not provide any additional assurance that the functions are more reliable. Considering this, the proposed amendment does not significantly reduce the margin of safety.

Part B—Trip System Logic for Physics Testing

During the low power physics testing, implementation of the proposed amendment will result in one power range channel being in a bypassed state. In this configuration, there will be three available channels with a two-out-of-three logic required to actuate the neutron flux trip function. As required by LCO 3.1.10, the testing will be performed while the reactor is in Mode 2 and therefore, restricted by the Technical Specifications to a power level of less than or equal to 5 percent.

There are two power range control functions, rod control and steam generator level control. At the 5 percent or less power level, rod control is in manual and is not affected by the testing configuration. Steam generator level control is not affected since its input from the NIS channel connected to the Reactivity Computer is placed in bypass when establishing the test configuration. Therefore, an assumed failure affecting these control functions does not have to be considered for the testing configuration. Also while in this configuration, an assumed single failure will not prevent the power range monitors from actuating as designed.

The reactor trip function of the intermediate range detectors will be unaffected by the proposed amendment and therefore, will be available to mitigate a reactivity transient at low power. Further, the trip setpoint for the power range monitors are decreased during startup of the reactor from the normal 109% setpoint to a value less than or equal to 85%. This setpoint reduction provides an additional measure to limit a reactivity excursion.

Based on the preceding, TVA concludes that there is no significant reduction in the margin of safety due to the implementation of the proposed amendment.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: June 22, 2000.

Description of amendment request: The proposed amendments to the Technical Specification Figures 3.4–2, 3.4–3, and associated Bases would extend the cumulative core burnup applicability limits for the reactor coolant system pressure-temperature (P/T) operating limits, Low Temperature Overpressure Protection System (LTOPS) setpoints, and LTOPS enable temperature (T enable). Implementation of American Society of Mechanical Engineers (ASME) Section XI Code Cases N–640 and N–514 will require exemptions from the requirements of 10 CFR 50, Appendix G.

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

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

The proposed changes extend the cumulative core burnup applicability of the existing North Anna Units 1 and 2 P/T limits, LTOPS setpoints, and T enable values. No changes to plant systems, structures, or components are proposed, and no new allowable operating modes are established, The P/T limits, LTOPS setpoints, and T enable values do not contribute to the probability of occurrence or consequences of accidents previously analyzed. The revised licensing basis analyses utilize acceptable analytical methods, and continue to demonstrate that established accident analysis acceptance criteria are met. Therefore, there is no increase in the probability or consequences of any accident previously evaluated.

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

The proposed changes extend the cumulative core burnup applicability of the existing North Anna Units 1 and 2 P/T limits, LTOPS setpoints, and T enable values. No changes to plant systems, structures, or components are proposed, and no new allowable operating modes are established. Therefore, the proposed changes do not create the possibility of any accident or malfunction of a different type previously evaluated.

3. Does the change involve a significant reduction in the margin of safety[?]

The proposed revised analysis bases use the ASME Section XI code Case N–640 K1c stress intensity formulation and a plant specific application of the analysis methodology which supports ASME Section XI Code Case N–514. These analysis features are less restrictive than those associated with the existing analyses, but are conservative with respect to [those] established by ASME Section XI margins. The proposed revised analyses support continued use of the existing North Anna Units 1 and 2 Technical specification P/T limit curves, LTOPS setpoints, LTOPS enable temperatures for North Anna Units 1 and 2 cumulative core burnups up to 32.3 effective full power years (EFPY) and 34.3 EFPY, respectively. The analyses demonstrate that established analysis acceptance criteria continue to be met. Specifically, the existing P/T limit curves, LTOPS setpoints, and LTOPS T enable values provide acceptable margin to vessel fracture under both normal operation and LTOPS design basis (mass addition and heat addition) accident conditions. Therefore, the proposed changes do 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.

Attorney for licensee: Mr. Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendment request: June 22, 2000.

Description of amendment request: The proposed changes would modify Facility Operating Licenses NPF4 and NPF–7, along with the associated Bases, to permit the elimination of the assumed increase in the rod control cluster assembly (RCCA) drop time resulting from a concurrent trip and seismic event, when determining if the measured rod drop times meet the Technical Specifications limit of 2.7 seconds.

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[?]

Elimination of the assumed increase in the RCCA drop time resulting from a concurrent trip and seismic event when determining if the measured rod drop times, including measurement uncertainties, meet the accident analysis limit[,] does not contribute to the probability of previously analyzed accidents. The proposed change will not alter the limiting results of the safety analyses presented in Chapter 15 of the UFSAR [Updated Final Safety Analysis Report]. Although the proposed change eliminates an accident consideration that is currently addressed in the UFSAR accident analyses (*i.e.* any Chapter 15 accident with the effects of a concurrent seismic occurrence reflected in the RCCA drop time), there is no significant increae in the probability or consequences of any accident previously evaluated.

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

There are no modifications to the plant as a result of the changes. No new accident or event initiators are created by eliminating the assumed increase in the RCCA drop time resulting from a concurrent trip and seismic event. The proposed change will not alter the ability of the reactor protection and control system to perform their design functions or to meet the applicable criteria set forth in the IEEE [Institute of Electrical and Electronics Engineers] and ANSI [American National Standards Institute] standards and in 10 CFR 50 Appendix A. Therefore, the proposed changes do not create the possibility of any accident or malfunction of a different type previously evaluated.

3. Does the change involve a significant reduction in the margin of safety[?]

The proposed change will not alter the limiting results of the safety analyses presented in Chapter 15 of the UFSAR. Elimination of the assumed increase in the RCCA drop time resulting from a concurrent trip and seismic event when determining if the measured rod drop times, including measurement uncertainties, [meet] the accident analysis limit maintains adequate safety margin in the safety analysis. Therefore, the proposed change does not significantly reduce a margin of safety.

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

Attorney for licensee: Mr. Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendment request: March 29, 2000.

Description of amendment request: The proposed change would revise Technical Specification (TS) 3.19 and TS 4.1. The change would reflect two redundant trains of bottled air for the main control room (MCR), include remedial action statements for one train and two trains inoperable, eliminate the extension of 8 hours to 24 hours currently permitted by TS 3.19.B, add requirements for an inoperable control room pressure boundary, and include additional surveillance testing requirements. The TS 3.19 Basis and TS 4.1 Basis would be revised for consistency with the respective 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:

The proposed TS change includes train specific requirements, adds requirements for an inoperable control room pressure boundary, imposes additional surveillance testing requirements for the MCR bottled air system, and is consistent with the existing accident analyses. We have reviewed the proposed TS change relative to the requirements of 10 CFR 50.92 and determined that a significant hazards consideration is not involved. Specifically, operation of Surry Power Station with the proposed change will not:

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

The proposed change does not involve a physical modification and does not modify the design or operation of the MCR bottled air system or the plant. Since the MCR bottled air system functions to respond to—not prevent—an accident, the probability of occurrence of an accident is not affected. The elimination of the currently allowed extension of the remedial action time, the addition of train specific requirements and inoperable boundary requirements, and the imposition of additional surveillance testing requirements serve to ensure no increase in the consequences of an accident. Therefore, the proposed change does not significantly increase the probability of occurrence or the consequences of any previously analyzed accident.

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

The proposed change does not involve a physical modification and does not affect the design or operation of the MCR bottled air system or the plant. Consequently, no new or unique operational modes or accident precursors are introduced. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change does not involve a physical modification and does not modify the design or operation of the MCR bottled air system or the plant. The elimination of the currently allowed extension of the remedial action time, the addition of train specific requirements and inoperable boundary requirements, and the imposition of additional surveillance testing requirements serve to ensure the bottled air system's ability to pressurize the main control room for one hour following a design basis accident, which is consistent with the existing accident analyses. Therefore, the proposed change does 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Acting Section Chief: L. Raghavan.

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

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

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

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

Date of application for amendments: July 13, 2000.

Description of amendments request: Amend Technical Specification 3.7.5.c to allow an increase in the average essential raw cooling water supply header temperature from 84.5°F to 87°F until September 30, 2000.

Date of publication of individual notice in the **Federal Register:** July 20, 2000 (65 FR 45113).

Expiration date of individual notice: August 3, 2000.

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 electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of application for amendment: August 23, 1999, as supplemented January 8, 2000.

Brief description of amendment: The amendment deletes certain license conditions that are obsolete and no longer apply.

Date of issuance: July 24, 2000.

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

Amendment No.: 130.

Facility Operating License No. NPF–62: The amendment revised the License.

Date of initial notice in ***Federal Register:*** September 22, 1999 (64 FR 51346). The January 8, 2000, submittal identified an additional license condition that was no longer applicable and thus did not change the scope of the action noticed or alter the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 24, 2000.

No significant hazards consideration comments received: No.

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: August 2, 1999, as supplemented April 7 and July 5, 2000.

Brief description of amendment: This amendment revises Technical Specification 6.2.2.e, ''Administrative Controls—Unit Staff.'' The license requirements for operations management have been modified.

Date of issuance: July 19, 2000.

Effective date: July 19, 2000.

Amendment No.: 99.

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

Date of initial notice in ***Federal Register:*** August 25, 1999 (64 FR 46426). The supplemental letters dated April 7 and July 5, 2000, contained clarifying information only, and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 19, 2000.

No significant hazards consideration comments received: No.

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: April 12, 2000, as supplemented June 2, 2000.

Brief description of amendment: This amendment revises Technical Specification (TS) 3/4.4.9.2, ''Pressure/Temperature (P-T) Limits—Reactor Coolant System,'' and TS 3/4.4.9.4, ''Overpressure Protection System,'' and the associated Bases. Specifically, the amendment incorporates results of the Reactor Vessel Surveillance Program capsule analysis and an exemption from 10 CFR 50.60(a), based on American Society of Mechanical Engineers Code Case N–640.

Date of issuance: July 28, 2000.
Effective date: July 28, 2000.
Amendment No. 100.
Facility Operating License No. NPF–63. Amendment revises the Technical Specifications.
Date of initial notice in ***Federal Register:*** May 3, 2000 (65 FR 25762). The supplemental letter dated June 2, 2000, contained clarifying information only, and did not change the initial no significant hazards consideration determination.
The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 28, 2000.
No significant hazards consideration comments received: No.

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

Date of amendment request: July 29, 1999, as supplemented by letters dated August 24, 1999, January 27, 2000, May 22, 2000, and May 31, 2000.
On June 14, 2000, the Commission published in the **Federal Register** (FR) Notice of consideration of issuance of amendment to facility operating license, proposed no significant hazards consideration determination, and opportunity for a hearing (65 FR 37425). In this finding, incorrect reference is made to supplements dated August 8, 1999, and March 29, 2000. No supplements from the licensee with these dates are related to this amendment.
Brief description of amendment: The amendment modifies Technical Specification 3.8.1.1 and associated Bases by extending the Emergency Diesel Generator (EDG) allowed outage time from 72 hours to ten days. In the supplemental letter dated May 22, 2000, an alternate source for the onsite power system during the EDG maintenance outage, by way of a temporary EDG (TEDG), was added. The application dated July 29, 1999, did not include the TEDG.
Date of issuance: July 21, 2000.
Effective date: As of the date of issuance and shall be implemented 60 days from the date of issuance.
Amendment No.: 166.
Facility Operating License No. NPF–38: The amendment revised the Technical Specifications.
Date of initial notice in ***Federal Register:*** June 14, 2000 (65 FR 37425). This notice is based on the supplement dated May 22, 2000, and supercedes the notice dated February 9, 2000 (65 FR 6406), which is based on the licensee's letter dated July 29, 1999. The May 31, 2000, supplement did not expand the scope of the application as noticed or change the proposed no significant hazards consideration determination.
The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 21, 2000.
No significant hazards consideration comments received: No

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

Date of application for amendment: March 6, 2000.
Brief description of amendment: Revised the Improved Technical Specification Action Condition and Surveillance Requirement related to the diesel-driven emergency feedwater pump (EFW–3) required lube oil volume.
Date of issuance: July 17, 2000.
Effective date: July 17, 2000.
Amendment No.: 192.
Facility Operating License No. DPR–72: Amendment revised the Technical Specifications.
Date of initial notice in ***Federal Register:*** April 19, 2000 (65 FR 21036).
The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 17, 2000.
No significant hazards consideration comments received: No.

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

Date of application for amendments: November 30, 1999, as supplemented March 8, May 15, and July 5, 2000.
Brief description of amendments: The proposed amendments would revise the Technical Specifications to allow the use of credit for soluble boron in the spent fuel pool criticality analyses. In addition, a revised criticality analysis for the fresh fuel storage racks will be used to update the licensing bases.
Date of issuance: July 19, 2000.
Effective date: July 19, 2000.
Amendment Nos.: 206 and 200.
Facility Operating License Nos. DPR–31 and DPR–41: Amendments revised the Technical Specifications.
Date of initial notice in ***Federal Register:*** May 3, 2000 (65 FR 25765). The May 15, and July 5, 2000, submittals provided clarifying information that did not change the scope of the original request or change the initial proposed no significant hazards consideration determination.
The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 19, 2000.
No significant hazards consideration comments received: No.

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

Date of application for amendments: April 27, 2000.
Brief description of amendments: Incorporate references to the NRC safety evaluations supporting exemptions granted for the Thermo-Lag Upgrade project. In addition, the amendments modify Technical Specification Section 6.0, Administrative Controls, Section 4.7.6.g, to include page 3/4 7–21 which was inadvertently excluded from the previous submittal and amendment.
Date of issuance: July 20, 2000.
Effective date: July 20, 2000.
Amendment Nos.: 207 and 201.
Facility Operating License Nos. DPR–31 and DPR–41: Amendments revised the Technical Specifications and the Operating Licenses.
Date of initial notice in ***Federal Register:*** May 31, 2000 (65 FR 34746).
The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 20, 2000.
No significant hazards consideration comments received: No.

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

Date of application for amendment: April 15, 1999, as supplemented December 22, 1999, and February 24, 2000.
Brief description of amendment: The amendment editorially revised the Technical Specifications to enhance clarity.
Date of Issuance: July 17, 2000.
Effective date: July 17, 2000 and shall be implemented within 30 days of issuance.
Amendment No.: 211.
Facility Operating License No. DPR–16: Amendment revised the Technical Specifications.
Date of initial notice in ***Federal Register:*** March 8, 2000 (65 FR 12293).
The February 24, 2000, supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.
The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated July 17, 2000.
No significant hazards consideration comments received: No.

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

Date of application for amendment: June 3, 1999, as supplemented on December 22, 1999.

Brief description of amendment: The proposed amendment revised the Technical Specifications to permit continued plant operation with a maximum of two inoperable recirculation loops, provided certain conditions are met. Oyster Creek's Technical Specifications (TSs), Section 3.3.F.2 currently permit operation with 4 of the 5 recirculation loops with certain constraints. If only 3 loops are operable, however, the TSs require plant shutdown within 12 hours. Analysis indicates that the plant may be safely operated at 90 percent power with three operable recirculation loops.

Two definitions are added to Section 1 of the TSs to specify the difference between an idle recirculation loop and an isolated recirculation loop. These definitions have been incorporated into the specification to provide an explicit description of acceptable valve configurations. In addition, several paragraphs have been added to the Bases of Section 3.3 and one paragraph in the Bases of Section 3.10 has been modified. In each case the Bases section has been segmented from the specification, which affects the pagination of the Bases.

Date of Issuance: July 27, 2000.

Effective date: July 27, 2000 and shall be implemented within 30 days of issuance.

Amendment No.: 212.

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

*Date of initial notice in **Federal Register:*** May 3, 2000 (65 FR 25766). The December 22, 1999, supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated July 27, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 11, 1997, as supplemented by letter dated May 8, 2000.

Brief description of amendment: The amendment revised the technical specifications by adding a new limiting condition for operation (LCO) for an inoperable engineered safety features logic subsystem. In addition, administrative changes were made to either support the new LCO or clarify existing text.

Date of issuance: July 25, 2000.

Effective date: July 25, 2000, and shall be implemented within 60 days from the date of issuance.

Amendment No.: 194.

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

*Date of initial notice in **Federal Register:*** February 11, 1998 (63 FR 6987). The May 8, 2000, supplemental letter provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 25, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 10, 2000.

Brief description of amendment: This amendment modifies the requirements contained in the Salem Unit No. 2 Technical Specifications regarding the operation of the movable incore detector system and allows continued operation of Salem Unit No. 2 through the remainder of Cycle 11. The revision represents a one-time change to allow use of the movable incore detector system for measurement of core peaking factors with less than 75% and greater than or equal to 50% of the detector thimbles available. Public Service Electric and Gas Company submitted this request in response to degradation of the movable incore detector system.

Date of issuance: July 25, 2000.

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

Amendment No.: 212.

Facility Operating License No. DPR–75: This amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 23, 2000 (65 FR 33378).

The Commission received comments which were addressed in the NRC staff's Safety Evaluation dated July 25, 2000.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 25, 2000.

No significant hazards consideration comments received: Yes.

Southern California Edison Company, et al., Docket Nos. 50–206, San Onofre Nuclear Generating Station (SONGS), Unit 1, San Diego County, California

Date of application for amendment: December 2, 1999, as supplemented on May 16, 2000.

Brief description of amendment: The amendment revised the SONGS Unit 1 Technical Specifications by revising the administrative controls to be consistent with the SONGS Unit 2 and 3 Technical Specification administrative controls including changes to the administrative control of working hours and working hour deviation approvals, position titles and responsibilities and organizational description reference, qualifications for a multi-discipline supervisor, quality assurance program control of review and audit and record retention procedures, high radiation area controls, description of the plant configuration for environmental protection, and environmental protection related document reporting. The amendment also incorporated changes related to certified fuel handlers and 10 CFR 50.54(x).

Date of issuance: July 19, 2000.

Effective date: July 19, 2000, to be implemented within 30 days of issuance.

Amendment No.: 159.

Facility Operating License No. DPR–13: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 29, 1999 (64 FR 73096).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 19, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 17, 1999, as supplemented on June 30, 2000.

Brief description of amendments: Revises License Condition to allow storage at the Sequoyah Nuclear Plant site of low-level radioactive waste generated at Watts Bar Nuclear Plant, Unit 1.

Date of issuance: July 18, 2000.

Effective date: July 18, 2000.

Amendment Nos.: 257 and 248.

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

*Date of initial notice in **Federal Register:*** February 23, 2000 (65 FR 9012). The supplemental letter of June 30, 2000, did not change the initial No Significant Hazards Consideration determination.

The Commission's related evaluation of the amendment is contained in an Environmental Assessment dated June 29, 2000 (65 FR 41739) and in a Safety Evaluation dated July 18, 2000.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–390, Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of application for amendment: November 20, 1998, as supplemented July 19, 1999, and January 21, 2000.

Brief description of amendment: The amendment revises the Technical Specifications (TS) to change the surveillance requirements for an inspection of the ice condenser flow channels that previously used a 0.38 inch ice/frost buildup criterion to a criterion that limits flow blockage to the 15 percent value that was used in the accident analysis. Changes to the Bases were also made. Tennessee Valley Authority also indicated that its proposal is consistent with TS Traveler Form No. 336.

Date of issuance: July 17, 2000.

Effective date: July 17, 2000.

Amendment No.: 25.

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

Date of initial notice in ***Federal Register:*** December 15, 1999 (64 FR 70093). The January 21, 2000, letter contained clarifying information that did not change the initial No Significant Hazards Consideration Determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 17, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 23, 2000.

Brief description of amendment: The amendment relocates the specifications for reactor coolant conductivity and chloride concentration from the Technical Specifications to the Technical Requirements Manual.

Date of Issuance: July 18, 2000.

Effective date: As of its date of issuance, and shall be implemented within 60 days of issuance.

Amendment No.: 190.

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

Date of initial notice in ***Federal Register:*** June 14, 2000 (65 FR 37430).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated July 18, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 23, 2000.

Brief description of amendment: The amendment revises the Technical Specifications to increase the interval between Local Power Range Monitor calibrations from 1,000 equivalent full power hours to 2,000 megawatt-days/ton.

Date of Issuance: July 18, 2000.

Effective date: As of its date of issuance, and shall be implemented within 60 days of issuance.

Amendment No.: 191.

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

Date of initial notice in ***Federal Register:*** June 14, 2000 (65 FR 37431).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated July 18, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 22, 2000.

Brief description of amendment: The amendment removes the Technical Specifications surveillance requirement for visual inspection of suppression chamber coating integrity once each refueling outage.

Date of Issuance: July 19, 2000.

Effective date: As of its date of issuance, and shall be implemented within 60 days.

Amendment No.: 192.

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

Date of initial notice in ***Federal Register:*** June 14, 2000 (65 FR 37430).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated July 19, 2000.

No significant hazards consideration comments received: No.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: March 31, 2000, as supplemented by letter of July 7, 2000.

Brief description of amendment: The amendment modifies the actions for Limiting Condition for Operation (LCO) 3.7.9, ''Ultimate Heat Sink (UHS),'' of the TSs. The new Action A for the LCO allows the plant to operate with the plant inlet water temperature of the UHS above 90°F, if the required lake water level is verified within 1 hour and once per 12 hours thereafter, but would require that the plant be shut down if the water temperature exceeded 94°F. The amendment replaces the requirement to shut down the plant if the UHS water temperature exceeds 90°F.

Date of issuance: July 14, 2000.

Effective date: July 14, 2000, to be implemented within 30 days from the date of issuance.

Amendment No.: 134.

Facility Operating License No. NPF–42. The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 19, 2000 (65 FR 21040). The supplemental letter of July 7, 2000, had minor clarifications that are within the scope of the initial notice and does not alter the no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 14, 2000.

No significant hazards consideration comments received: No.

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

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

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

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

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

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

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

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

For further details with respect to the action see (1) the application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By September 8, 2000, 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 electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

Date of application for amendment: July 13, 2000, as supplemented by letters dated July 14 and 21, 2000.

Brief description of amendment: The amendment permitted a one-time change to Technical Specification 4.4.5.0 and allowed alternate inspection scope and expansion criteria for steam generator tube inspections to be implemented during the mid-cycle outage scheduled for summer 2000.

Date of issuance: July 26, 2000.

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

Amendment No.: 217.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: Yes.

The NRC published a public notice of the proposed amendment, issued a proposed finding of no significant hazards consideration, and requested that any comments on the proposed no significant hazards consideration be provided to the staff by the close of business on July 24, 2000. The notice was published in The Courier (in Russellville) and the Arkansas Democrat-Gazette (in Little Rock) from July 20 through 22, 2000. No public comments were received.

The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of Arkansas, and final no significant hazards consideration determination are contained in a Safety Evaluation dated July 26, 2000.

Dated at Rockville, Maryland, this 3rd day of August 2000.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

Director, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 00–20014 Filed 8–8–00; 8:45 am]

BILLING CODE 7590–01–P

OCCUPATIONAL SAFETY AND HEALTH REVIEW COMMISSION

Submission for OMB Review; Comment Request

AGENCY: Occupational Safety and Health Review Commission (OSHRC)

SUMMARY: The Executive Director, OSHRC invites comments on the submission for OMB review as required by the Paperwork Reduction Act of 1995.

DATES: Interested persons are invited to submit comments on or before September 7, 2000.

ADDRESSES: Written comments should be addressed to Stuart Shapiro, Office of Management and Budget, Room 10202, New Executive Office Building, Washington, D.C. 20503 or should be electronically mailed to the internet address Stuart__Shapiro@omb.eop.gov.

SUPLEMENTARY INFORMATION: Section 3506 of the Paperwork Reduction Act of 1995 (44 U.S.C. requires that the Office of Management and Budget (OMB) provide interested Federal agencies and the public opportunity to comment on information collection requests. OMB may amend or waive the requirement for public consultation to the extent that public participation in the approval process would defeat the purpose of the information collection, violate State or Federal law, or substantially interfere with any agency's ability to perform its statutory obligations. The Executive Director published a notice containing proposed information collection request in the **Federal Register** date May 31, 2000. The proposed information collection included: (1) Type of review requested, (2) Title, (3) Summary of the collection, (4) Description of the need for, and proposed use of, the information, (5) Respondents and frequency of collection, and (6) Reporting and recordkeeping burden. OMB invites public comment.

Dated: August 3, 2000.

Patricia A. Randle,

Executive Director, Occupational Safety and Health Administration.

Type of Review: New.

Title: Evaluation of ''E–Z Trial.

OMB Number: New.

Frequency: Once.

Affected Public: Employers and/or their representatives, and labor organizations who have been involved in cases with the Review Commission.

Reporting and Recordkeeping Hour Burden:

Responses: 100

Burden hours: 75

Abstract: The Occupational Safety and Health Review Commission (OSHRC) published a rule in the **Federal Register** dated August 14, 1995 establishing the ''E–Z Trial'' program. The rule was subsequently amended to eliminate the sunset provisions in the original rule and to revise the procedural rules governing the ''E–Z Trial'' program effective July 31, 1997. We are evaluating the program as modified effective July 31, 1997. The evaluation will involve surveying employers and employer representatives regarding their satisfaction with the fairness and efficiency of the process. The evaluation will also analyze data on the rate at which ''E–Z Trial'' cases go to a hearing, and on the length and cost of hearings. Finally, the evaluation will study the cycle times of these cases as compared to those of conventional cases. Information will also be gathered from Occupational Safety and Health Administration (OSHA) staff and from the Solicitor of Labor.

[FR Doc. 00–20080 Filed 8–8–00; 8:45 am]

BILLING CODE 7600–01–P

SECURITIES AND EXCHANGE COMMISSION

[Release No. 24590; 812–12064]

Investment Company Act; Hillview Investment Trust II, et al.; Notice of Application

August 3, 2000.

AGENCY: Securities and Exchange Commission (''Commission'').

ACTION: Notice of an application under section 17(b) of the Investment Company Act of 1940 (the ''Act'') for an exemption from section 17(a) of the Act.

SUMMARY OF APPLICATION: Applicants request an order to permit a limited partnership to transfer all of its assets to a corresponding new series of a