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 April 8, 2000, through April 21, 2000. The last biweekly notice was published on April 19, 2000 (65 FR 21034).

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 June 2, 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, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: April 7, 2000.

Description of amendment request: The proposed amendment would revise Harris Nuclear Plant (HNP) Technical Specification (TS) 3/4.7.6, ''Control Room Emergency Filtration System,'' TS 3/4.7.7, ''Reactor Auxiliary Building Emergency Exhaust System,'' TS 3/4.9.12, ''Fuel Handling Building Emergency Exhaust System,'' and the associated Bases. Specifically, the licensee proposes to revise these TS to provide an Action when the Control Room Emergency Filtration System or Reactor Auxiliary Building Emergency Exhaust System ventilation boundary is inoperable and a note that allows an applicable ventilation boundary to be open intermittently under administrative controls. Additionally, the licensee proposes to modify TS 3/4.3.3.1, ''Radiation Monitoring for Plant Operations,'' to provide consistency between the applicability of the Control Room Emergency Filtration System and the radiation monitors that initiate a Control Room Isolation signal.

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

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

Ventilation systems are not accident initiating systems as described in the Final Safety Analysis Report. The changes are based on the low probability of a design basis accident occurring during the 24 hour completion time and compensatory measures available to minimize dose consequences of an event during this time. The proposed change does not affect another Structure, System, or Component.

Current HNP TS do not restrict fuel movement in the fuel handling or loads over spent fuel pools concurrent with an inoperable Control Room Emergency Filtration System. Providing restrictions for fuel movement and loads over spent fuel pools preserves assumptions made in the fuel handling accident analysis. The addition of applicability requirements for fuel movement and movement of loads over spent fuel pools is consistent with NUREG–1431, Revision 1, and is more restrictive than current HNP TS.

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

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

Ventilation systems are not accident initiating systems as described in the Final Safety Analysis Report. As such, the failure of the ventilation system to operate properly or a premature actuation of the ventilation system can not initiate an accident.

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

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

The proposed change to ventilation systems does not significantly affect any of the parameters that relate to the margin of safety as described in the Bases of the TS or the FSAR [Final Safety Analysis Report]. Accordingly, NRC Acceptance Limits are not affected by this change. The changes are based on the low probability of a design basis accident occurring during the 24 hour completion time and compensatory measures available to minimize dose consequences of an event during this time.

The addition of applicability requirements for Control Room Emergency Filtration System during movement of irradiated fuel assemblies and movement loads over spent fuel pools provide additional margin not currently provided in HNP TS.

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

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

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.

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

Date of amendment request: April 12, 2000.

Description of amendment request: The proposed amendment would revise Harris Nuclear Plant (HNP) 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 licensee proposes to revise the applicable TS to incorporate results of the Reactor Vessel Surveillance Program capsule analysis. A summary report was previously submitted to the NRC (HNP–99–157, dated 11/9/99) in accordance with Title 10 of the Code of Federal Regulations, Part 50 (10 CFR 50), Appendix H. Additionally, the licensee's submittal requested an exemption to 10 CFR 50.60 (a), based on American Society of Mechanical Engineers (ASME) Code Case N–640 and WCAP–15315. The exemption request will be evaluated separate from the proposed license amendment.

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

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

The proposed changes affect operations of the Reactor Coolant System (RCS) components when the RCS temperature is below 350° F. The revisions to P–T limits and allowable heatup and cooldown rate limits are consistent with ASME code cases which have been authorized for other licensees by the NRC. The proposed changes modify the setpoint of the pressurizer PORVS [power operated relief valves] for LTOPS [low temperature overpressure setpoints]. Changes to the LTOPS setpoints applicable below 350° F effectively increase the allowable operating pressure for any given temperature during shutdown. These changes do not result in conditions which are outside of the design basis for RCS Structures, Systems, and Components (SSCs). Therefore, the proposed changes do not alter the characteristics of the RCS SSCs adversely, and therefore do not impact the performance of the RCS SSCs during power operations.

The revised P–T limits and heatup and cooldown rate limits are within the design capabilities of the RCS SSCs and pressure control systems. While the proposed new P–T limits are less restrictive than the current Technical [Specification] requirements, they assure that plant operation is within the design capacity of the reactor vessel materials. Therefore, the RCS capability as a fission product barrier is not compromised.

The changes to the LTOPS setpoints do not affect accident consequences since no credit is assumed for operation of LTOPS to mitigate accidents.

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

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

The proposed changes do not involve new plant components or procedures, but only revise existing operational limits and setpoints. These changes do not place SSCs in conditions outside of their design basis, and the revised operating setpoints and conditions are within the capability of the plant control systems.

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

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

The proposed changes to the P–T limits and LTOPS setpoints change the calculational method from that described in the bases to one based on ASME Code Case N–640, and on WCAP–15315. The effect of this change is to allow plant operation with different limits, but still with adequate margins to assure the integrity of the reactor vessel and RCS.

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

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

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

Date of amendment request: March 15, 2000.

Description of amendment request: The proposed amendments would revise the ultimate heat sink temperature in the technical specifications from 98°F to 100°F.

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 any accident previously evaluated?

Analyzed accidents are assumed to be initiated by the failure of plant structures, systems or components. An inoperable Ultimate Heat Sink (UHS), which is the source of water for the Essential Service Water (ESW) System, is not considered as an initiator of any analyzed events. The analyses for Braidwood Station, Units 1 and 2, assume a UHS temperature of 100°F. Therefore, continued operation with a UHS temperature less than or equal to 100°F will not increase the probability of occurrence of any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). The proposed change does not involve any physical alteration of plant systems, structures or components. A UHS temperature of up to 100°F does not increase the failure rate of systems, structures or components because the systems, structures or components are rated and analyzed for operation with ESW temperatures of 100°F and the design allows for higher temperatures than at which they presently operate.

The basis provided in Regulatory Guide 1.27 "Ultimate Heat Sink for Nuclear Power Plants," Revision 2, dated January 1976, was employed for the temperature analysis of the Braidwood Station UHS to implement General Design Criteria (GDC) 44, "Cooling water," and GDC 2, "Design bases for protection against natural phenomena," of Appendix A to 10 CFR Part 50. This Regulatory Guide was employed for both the original design/licensing basis of the Braidwood Station UHS and a subsequent evaluation which investigated the potential for increasing the average water temperature of the UHS from ≤98°F to ≤100°F. The heat loads selected for the UHS analysis considered one Braidwood Station unit in a Loss of Coolant Accident (LOCA) condition concurrent with a Loss Of Offsite Power (LOOP) event and the remaining Braidwood Station unit undergoing a safe non-accident shutdown. In the analysis, these heat loads are removed by the UHS using only ESW pumps. The main cooling pond is conservatively assumed not to be available at the start of the event. The analysis shows that with an initial UHS temperature of 100°F, the required heat loads can be met for 30 days while maintaining ESW temperatures at acceptable values.

Based on the above, it has been demonstrated that the operation at an initial UHS temperature of ≦100° F at the start of the design basis event will result in the continued ability of the equipment and components supplied by the ESW system to perform their intended safety functions.

Therefore, increasing the average water temperature limit of the UHS from ≦98° F to ≦100° F does not increase the consequences of any accident previously evaluated. Raising this limit does not introduce any new equipment, equipment modifications, or any new or different modes of plant operation, nor does it affect the operational characteristics of any equipment or systems. Therefore, this proposed change does not involve a significant 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 change does not involve a physical alteration of the units. There is no change being made to the parameters within which the units are operated that is not bounded by the analyses. There are no setpoints at which protective or mitigative actions are initiated that are affected by this proposed change. This proposed change will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alteration in the procedures that ensure the units remain within analyzed limits is proposed, and no change is being made to procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced. The proposed change does not alter assumptions made in the safety analysis.

Increasing the allowed average water temperature of the UHS in Technical Specification (TS) 3.7.9, ''Ultimate Heat Sink (UHS),'' has no impact on plant operation. Operating at the proposed higher temperature limit does not introduce new failure mechanisms for systems, structures or components. The engineering analyses performed to support the change to UHS temperature limit provides the basis to conclude that the equipment is designed for operation at elevated temperatures. The current analyses and calculations assume a UHS temperature of 100° F, which is within the design limits of the affected equipment. In addition, design and construction codes applied to the affected structures, systems and components provided sufficient margin to accommodate the proposed temperature change.

Therefore, this proposed 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?

The proposed change allows operation with the UHS temperature ≦100° F. The margin of safety is determined by the design and qualification of the plant equipment, the operation of the plant within analyzed limits, and the point at which protective or mitigative actions are initiated. The proposed change does not impact these factors. The existing analyses already assume an initial UHS temperature of 100° F for design basis accident conditions. There are no required design changes or equipment performance parameter changes associated with this change. No protection setpoints are affected as a result of this change. This temperature increase has been confirmed to not change the operational characteristics of the design of any equipment or system. All accident analysis assumptions and conditions will continue to be met. Thus, the proposed increase in UHS temperature 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≤980767.

NRC Section Chief: Anthony J. Mendiola.

Detroit Edison Company, Docket No. 50–16, Enrico Fermi Atomic Power Plant, Unit 1, Monroe County, Michigan

Date of amendment request: February 11, 1999 (Reference NRC–00–0023).

Description of amendment request: The proposed amendment will revise the Technical Specifications by: (1) Deleting Specification A.8, the definition of ''Primary System'' which will no longer be necessary if the specifications related to the Primary System cover gas system are deleted; (2) deleting Specification D, which specifies the requirements for the Primary System cover gas system; (3) deleting the portion of Specification H.1 that specifies the surveillance requirements for the Primary System pressure alarms; (4) deleting Table H.1 item a, the Primary System pressure alarm points; (5) deleting Specification H.3.b, the requirement to perform surveillances of the door and seals around the machinery dome; (6) deleting Specification I.7.b, which requires procedures for maintaining cover gas supply; and (7) deleting Specification I.9.d, which requires keeping records of CO_2 cover gas usage. The above-listed changes would allow the licensee to remove the Primary System cover gas system from service, an action that would allow the licensee to begin work on removing the remaining residual sodium from the Primary System. The licensee also requested an editorial change in Table H.1 item b.1, to change ''Bldg.'' to ''Building''.

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 using the standards in 10 CFR 50.92(c). The licensee's analysis is presented below:

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

Removing the cover gas from the [P]rimary [S]ystem, opening the [P]rimary [S]ystem, and cleaning out sodium residues will not significantly increase the probability of an accident occurring as long as the probability of an uncontrolled water reaction with the sodium is not significantly increased. This is done by conducting cutting operations and sodium reactions under control conditions. Removing the cover gas or opening the system will not take place until the current asbestos abatement project in the Reactor Building is complete since water is being used. The abatement is expected to be completed this winter before the license amendment will be approved. Note that EPA approval for dry removal has been obtained for where there is a risk of water coming into contact with sodium. The successful dismantling of the secondary sodium system piping in the Steam Generator building demonstrates that sodium systems can be cut open safely. The sodium residue processing in the secondary sodium storage tanks demonstrates sodium cleanup can be conducted safely. The consequences of an accident will not be increased by removing the cover gas, opening the [P]rimary [S]ystem, or reacting the sodium residues because the previously analyzed accidents already involve the release of all the radioactive material in the [P]rimary [S]ystem and all the radioactive material in the liquid waste system. The maximum postulated dose to the public was analyzed to be within the 10 CFR [Part] 20 limit of 100 mrem/year. This change will not increase the amount of radioactive material available to be released.

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

Removing the cover gas from the [P]rimary [S]ystem, opening the [P]rimary [S]ystem, and cleaning out the sodium residues will not create a new or different type of accident. A sodium accident has been previously evaluated. The only other type of accident which could possibly be caused by removing the [P]rimary [S]ystem cover gas, opening the [P]rimary [S]ystem, or processing primary sodium residues is a liquid waste release, which is highly unlikely. A liquid waste accident has also been previously evaluated. Only the [P]rimary [S]ystem and other equipment or piping containing primary sodium is expected to be affected by this change.

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

Only a relatively small amount of sodium remains in the [P]rimary [S]ystem and other equipment containing primary sodium. Some of this residual may have been converted to sodium carbonate, leaving even less sodium remaining. The cover gas was a good precaution, especially for systems sitting unattended for many years. It prevented moisture from intruding into the systems and reacting with the sodium residues. It prevented oxygen from entering and reacting with any hydrogen formed from reactions of water and sodium. Discontinuing the use of cover gas slightly reduces the margin of safety, but not significantly. Removing the cover gas does not, in itself, introduce water into the system in an uncontrolled manner. Even if slight amounts of moisture from humidity in the air enter over the next year or two until the sodium is removed while the system is opened or unsealed, the system volume is large enough that the system will be able to dissipate any small reactions that occur. In addition, the calculated consequence[s] of releasing the radioactive material in the primary sodium is small and well within 10 CFR [Part] 20 and Technical Specification limits.

The planned processing of sodium residues is evaluated as releasing the radioactive material to the atmosphere, as planned release using controls specified in the Technical Specifications for gaseous effluents. For these reasons, the proposed change does not involve a significant reduction in the margin of safety.

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, NRC staff proposes to determine that the amendment request involves no significant hazards consideration. Attorney for licensee: John Flynn, Esquire, Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Branch Chief: Larry W. Camper.

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: November 30, 1999, as supplemented March 8, 2000.

Description of amendment request: 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. Criticality analyses were performed using the methodology developed by the Westinghouse Owners Group and described in WCAP–14416–NP–A, Revision 1, Westinghouse Spent Fuel Rack Criticality Analysis Methodology.

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.

There is no increase in the probability of a fuel assembly drop accident in the Spent Fuel Pool (SFP) when considering the presence of soluble boron in the SFP water for criticality control. The handling of the fuel assemblies in the SFP has always been performed in borated water. The consequences of a fuel assembly drop accident in the SFP are not affected when considering the presence of soluble boron.

There is no increase in the probability of the accidental misloading of spent fuel assemblies into the SFP racks when considering the presence of soluble boron in the pool water for criticality control. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures and will be in accordance with the Technical Specification (TS) spent fuel rack storage limitations. There is no increase in the consequences of the accidental misloading of spent fuel assemblies into the SFP racks because criticality analyses demonstrate that the pool will remain subcritical following an accidental misloading if the pool contains an adequate boron concentration. The proposed TS ensure that an adequate SFP boron concentration will be maintained. There is no increase in the probability of the loss of normal cooling to the SFP water when considering the presence of soluble boron in the pool water for subcriticality control since a high concentration of soluble boron has always been maintained in the SFP water.

A loss of normal cooling to the SFP water causes an increase in the temperature of the water passing through the stored fuel assemblies. This causes a decrease in water density, which would result in a net increase in reactivity when soluble boron is present in the water and Boraflex neutron absorber panels are present in the racks. However, the additional negative reactivity provided by the 1950 ppm boron concentration limit, above that provided by the concentration required (650 ppm) to maintain K_{eff} less than or equal to 0.95, will compensate for the increased reactivity which could result from a loss of SFP cooling event. Because adequate soluble boron will be maintained in the SFP water, the consequences of a loss of normal cooling to the SFP will not be increased.

The Fresh Fuel racks are analyzed by employing the ''Westinghouse Spent Fuel Rack Criticality Analysis Methodology'' approved by the NRC and described in WCAP–14416, NP–A, Revision 1. Only the method for Fresh Fuel storage racks criticality calculations has changed. The method of handling fuel, the maximum fuel enrichment, and the limiting values for criticality have not changed. Therefore, there is no change in the margin of safety for the Fresh Fuel storage racks.

Therefore, based on the conclusions of the above analysis, the proposed changes 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 amendments would not create the possibility of a new or different kind of accident from any previously evaluated.

Spent fuel handling accidents are not new or different types of accidents, they have been analyzed in Section 14.2.1 of the Updated Final Safety Analysis Report (UFSAR). Criticality accidents in the SFP are not new or different types of accidents, they have been analyzed in the UFSAR and in the spent fuel storage criticality analysis. Current TS 3/4.9.14 already contains a limit on the SFP boron concentration. The boron concentration in the SFP has always been maintained near the limit of the RWST boron concentration for refueling purposes. The current TS boron concentration requirement for the SFP water conservatively bounds the boration assumptions of the revised criticality analyses. Since soluble boron has always been maintained in the SFP water, the implementation of this requirement for criticality purposes will have no effect on normal pool operations and maintenance.

Since soluble boron has always been present in the SFP, a dilution of the SFP soluble boron has always been a possibility. However, it was shown in the SFP dilution analysis that a dilution of the Turkey Point SFP which could increase the spent fuel storage rack K_{eff} to greater than 0.95 is not a credible event. Therefore, the implementation of limitations on the SFP boron concentration for criticality purposes will not result in the possibility of a new or different kind of accident.

Proposed TS 3/4.9.14 Table 3.9–1 specifies the requirements for the spent fuel rack storage, which is currently contained in the TS. These proposed new SFP storage limitations are consistent with the assumptions made in the spent fuel rack criticality analysis, and will not have any significant effect on normal SFP operations and maintenance, and will not create any possibility of a new or different kind of accident. Verifications will continue to be performed to ensure that the SFP loading configuration meets specified requirements.

The Fresh Fuel racks are analyzed by employing the ''Westinghouse Spent Fuel Rack Criticality Analysis Methodology'' approved by the NRC and described in WCAP–14416, NP–A, Revision 1. Only the method for Fresh Fuel storage racks criticality calculations has changed. The method of handling fuel, the maximum fuel enrichment, and the limiting values for criticality have not changed. Therefore, there is no change in the margin of safety for the Fresh Fuel storage racks.

As discussed above, the proposed changes will not create the possibility of a new or different kind of accident from any previously evaluated. There is no significant change in plant configuration, equipment design or equipment.

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

The proposed TS changes will provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. Those limits are based on a plant specific criticality analyses performed in accordance with the NRC approved Westinghouse Spent Fuel Rack criticality analysis methodology.

The criticality analysis takes credit for soluble boron to ensure that K_{eff} will be less than or equal to 0.95 under normal circumstances. Storage configurations have been defined using a 95/95 K_{eff} calculation to ensure that the spent fuel rack K_{eff} will be less than 1.0 with no soluble boron. Soluble boron credit is used to provide safety margin by maintaining K_{eff} less than or equal to 0.95, including uncertainties, tolerances, and accident conditions in the presence of SFP soluble boron.

The loss of substantial amounts of soluble boron from the SFP that could lead to exceeding a K_{eff} of 0.95 has been evaluated in the SFP Dilution analysis and shown to be not credible.

The analysis shows that the dilution of the SFP boron concentration from 1950 ppm to 650 ppm is not credible. When this result is combined with the results from the 95/95 criticality analyses, which show that the spent fuel rack K_{eff} will remain less than 1.0 when flooded with unborated water, it provides a level of safety comparable to the conservative criticality analysis methodology required by ANSI 57.2–1983, NUREG–0800, and Regulatory Guide 1.13.

The Fresh Fuel racks are analyzed by employing the ''Westinghouse Spent Fuel Rack Criticality Analysis Methodology'' approved by the NRC and described in WCAP–14416, NP–A, Revision 1. Only the method for Fresh Fuel storage racks criticality calculations has changed. The method of handling fuel, the maximum fuel enrichment, and the limiting values for criticality have not changed. Therefore, there is no change in the margin of safety for the Fresh Fuel storage racks.

Therefore, the proposed changes in these license amendments will not result in a significant reduction in the plant's 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.

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

Date of amendment request: June 3, 1999, as supplemented on December 22, 1999

Description of amendment request: The proposed amendment would 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.

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

When operating with two inoperable recirculation loops, the proposed section 3.3.F.2.b requires that the reactor core thermal power not exceed 90% of rated power. This is a physical limitation of the plant conditions because maximum power is about 90% of rated power at the maximum recirculation flow with only three recirculation pumps operating. As such, the 90% of rated power becomes a limiting condition for three-loop operation. The licensee states that the results of this analysis conform to all the requirements of 10 CFR 50.46 and Appendix K.

The licensee analyzed recirculation pump trip transients for single and multiple pump trips. Although the transient in general is very mild, the licensee considers the case of simultaneous trip of all five pumps to be the limiting event among all possible recirculation pump trip events. For three-loop operation, given the requirement that the power level be maintained at or below 90% of rated power, the transient resulting from the loss of all three pumps would be bounded by the five-pump-trip event.

The proposed change, which permits three loop operation with a maximum of two idle or one idle and one fully isolated loop, will provide adequate safety margins during transient and accident conditions. The proposed changes do not affect any accident precursors because the accident occurrence is not dependent on the number of operating recirculation loops. Therefore, the probability of an accident previously evaluated is not increased. The proposed TS change will assure the ability of systems to perform their intended function. Therefore, the proposed changes will not introduce a significant increase in the consequences of an accident previously evaluated. Therefore, the probability of occurrence or the consequences of an accident previously evaluated in the Safety Analysis Report (SAR) will not increase as a result of these changes.

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

The proposed changes permit three-loop operation with a maximum of two idle or one idle and one fully isolated loop. The licensee considers the case of simultaneous trip of all the five pumps to be the limiting event among all possible recirculation pump trip events. For three-loop operation, given the requirement that the power level be maintained at or below 90% of rated power, the transient resulting from the loss of all three pumps would be bounded by the five-pump-trip event.

The proposed changes will not create a possibility for an accident or transient of a different type than any previously identified in the SAR.

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

The proposed changes will not decrease the margin of safety as defined in the basis of any Technical Specification. All relevant transient and accident scenarios have been analyzed for the conditions of three-loop operation and have demonstrated adequate margin to safety limits. Therefore, the proposed changes do not involve a significant reduction in the margin of safety. They neither adversely affect the performance characteristics of systems nor do they affect the ability of systems to perform their intended function. Therefore, the proposed changes do 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: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: M. Gamberoni, Acting.

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

Date of amendment request: March 17, 2000.

Description of amendment request: This amendment request proposes to revise the Cooper Nuclear Station Technical Specifications to incorporate the recommended Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal,'' laboratory testing protocol of American Society for Testing and Materials (ASTM) D3803–1989 for Engineered Safety Feature ventilation system charcoal samples.

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

The proposed charcoal testing changes and explicit reference to American Society for Testing and Materials (ASTM) D3803–1989 nuclear-grade activated charcoal test protocol do not affect Engineered Safety Feature (ESF) ventilation system operation or performance, reliability, actuation setpoints, or accident mitigation capabilities. The proposed changes also do not affect the operation and performance of any other equipment important to safety at Cooper Nuclear Station (CNS). ASTM D3803–1989 is a more accurate and demanding test which ensures that the charcoal filter efficiencies assumed in the CNS accident dose analysis are maintained. The proposed changes involve ESF ventilation system charcoal testing only and do not affect accident initiators. Therefore the proposed changes do not significantly increase the probability or consequences of an accident previously evaluated, as revised by the design basis accident radiological assessment calculational methodology revision submitted to the NRC under Reference 3 [in the March 17, 2000, amendment request].

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

The charcoal testing changes, and explicit reference to ASTM D3803–1989 nuclear-grade activated charcoal test protocol, do not affect ESF ventilation system operation or performance, or the operation and performance of any other equipment important to safety at CNS. The proposed changes clarify and explicitly identify the testing of the ESF ventilation system charcoal samples. No new or different accident scenarios, transient precursors, failure mechanisms, plant operating modes, or limiting single failures are introduced as a result of these changes. Therefore, the possibility of a new or different kind of accident from that previously evaluated, as revised by the design basis accident radiological assessment calculational methodology revision submitted to the NRC under Reference 3, is not created by this change.

3. Does not create a significant reduction in the margin of safety.

The required performance of the ESF ventilation systems following a design basis accident is not impacted by utilizing a more demanding protocol for charcoal testing. Thus, the margin of safety assumed in the CNS accident analysis, as revised by the design basis accident radiological assessment calculational methodology revision submitted to the NRC under Reference 3, is maintained. Revising the Technical Specifications to clarify charcoal testing methodology and explicitly referencing the charcoal [adsorber] testing being performed does not affect ESF ventilation system performance or operation, or the operation and performance of any other equipment important to safety at CNS. Therefore, these changes do not result in a significant reduction in any margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment requests: December 23, 1999.

Description of amendment requests: The proposed amendment would revise improved TS (ITS) 5.5.9.d.1.j)(iv) to change the tube support plate (TSP) intersections that are excluded from application of steam generator (SG) tube voltage based repair criteria for outside diameter stress corrosion cracking indications at TSPs.

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.

Application of a smaller wedge region exclusion zone [due to loss of coolant accident (LOCA) plus safe shutdown earthquake (SSE)] and a new seventh tube support plate (TSP) bending stress exclusion zone (due to feedline break (FLB)/steamline break (SLB) plus SSE) with respect to alternate repair criteria (ARC), does not increase the probability of tube burst or leakage following a postulated main steam line break (MSLB). Exclusion zones tubes will be inspected by bobbin every outage and by rotating pancake coil (RPC) if bobbin detects degradation. Tubes containing RPC-confirmed crack-like degradation at wedge region exclusion zone intersections and at the seventh TSP bending exclusion zone intersections will be plugged.

Tube burst criteria are inherently satisfied during normal operating conditions because of the proximity of the TSP. It is conservatively assumed that the entire crevice region is uncovered because of TSP displacement during the secondary side blowdown of a MSLB. Therefore, during a postulated MSLB accident, tube burst capability must exceed the Regulatory Guide 1.121 criterion requiring a margin of 1.43 times the SLB pressure differential on tube burst.

Relative to the expected leakage during accident condition loadings, a postulated MSLB outside of containment, but upstream of the main steam isolation valve, represents the most limiting radiological condition. The steam generator (SG) tubes are subjected to an increase in differential pressure following a MSLB, resulting in a postulated increase in leakage and associated offsite doses. Leakage following a MSLB bypasses containment.

Following each inspection, condition monitoring will be performed to verify that tube burst and leakage performance criteria were satisfied for all degradation.

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

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

Implementation of revised ARC exclusion zones does not introduce any significant change to the plant design basis. Use of new exclusion zones does not create a mechanism which could result in an accident in the free span. It is expected that for all plant conditions, neither a single nor multiple tube rupture event would likely occur in a SG where ARC exclusion zones have been applied.

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

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

Revised wedge region exclusion zones are based on a DCPP-specific analysis under locked tube conditions for the combined effects of a LOCA and SSE. The number of wedge region tubes that are predicted to collapse has been decreased when compared to the prior analysis, which used highly conservative assumptions. The revised analysis incorporates DCPP-specific LOCA and seismic loads that were not available when the prior analysis was performed. However, the revised analysis also yields conservative results, such that the number of tubes in the exclusion zone (244 per SG) bound the number of tubes calculated to collapse (144 per SG). Tubes located in the revised wedge region exclusion zone will continue to be subject to enhanced eddy current inspection requirements and will be excluded from application of ARC. Thus, existing tube integrity requirements apply to these tubes and the margin of safety is not reduced.

New seventh TSP bending exclusion zones are also based on a DCPP-specific analysis under locked tube conditions for the combined effects of a FLB/SLB and SSE. The analysis yields conservative results, such that 914 tubes per SG at the seventh TSP are assumed to exceed the Westinghouse lower tolerance limit yield stress of the tubing. Tubes located in the seventh TSP bending exclusion zone will be subject to enhanced eddy current inspection requirements and will be excluded from application of ARC. Thus, existing tube integrity requirements apply to these tubes and the margin of safety is not reduced.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Christopher J. Warner, Esq., Pacific Gas and Electric

Company, P.O. Box 7442, San Francisco, California 94120.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: April 6, 2000.

Description of amendment request: The Virgil C. Summer Nuclear Station Technical Specifications are being revised to change the definitions and surveillance requirements for response time testing of the Engineered Safety Feature Actuation System (ESFAS) and the Reactor Trip System (RTS). These changes will permit the verification of response time, whereas the current definitions imply the response time must be measured.

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?

This change to the Technical specifications (TS) 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 and ESFAS instrumentation is being used; the time response allocations/modeling assumptions in the Final Safety Analysis Report (FSAR) Chapter 15 analyses are still the same; only the method of verifying the 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 change will not change, degrade or prevent actions or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the FSAR.

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

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

This change does not alter the performance of process protection racks, Nuclear Instrumentation, and logic systems used in the plant protection systems. These systems will still have response time verified by test before being placed in operational service. Changing the method of periodically verifying instrument[ation] for these systems (assuring equipment operability) from response time testing to calibration and channel checks will not create any new accident initiators or scenarios. Periodic surveillance of these systems will continue and may be used to detect degradation that could cause the response time to exceed the total allowance. The total time response allowance for each function bounds all degradation that cannot be detected by periodic surveillance. Implementation of the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in 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 the process protection racks, Nuclear Instrumentation, and logic systems is modified to allow the 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 might cause the system response time to exceed the total allowance. The total response time allowance for each function bounds all degradation that cannot be detected by periodic surveillance. Based on the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety.

Pursuant to 10 CFR 50.91, the preceding analyses provides a determination that the proposed Technical Specifications change poses no significant hazard as delineated by 10 CFR 50.92.

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

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

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: April 6, 2000.

Description of amendment request: The proposed Technical Specifications change request (TSCR) seeks to remove the prescriptive testing requirements of TS 4.8.1.1.2.i.2 to allow the ASME Code Class 3 portions of the diesel fuel oil system to be pressure tested in accordance with Section XI of the ASME Boiler and Pressure Vessel Code as required by TS 4.0.5. This will permit the use of Code Case N–498–1 as accepted by Regulatory Guide 1.147, Revision 12, for assessment of the diesel fuel oil system pressure boundary integrity.

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

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

Industry experience has shown that an inservice leak test conducted at normal operating temperature and pressure is just as effective at finding leakage as a hydrostatic test conducted at 110% of the design pressure.

Therefore, there is no increase in the probability or consequences of previously evaluated accidents.

Also note that the Diesel Generator Fuel Oil System is not specifically modeled in the VCSNS Probability Risk Assessment. It is contained in the diesel generator fail to run event that has a probability of 5.8E–2. If the diesel generator fuel oil system had been modeled, pipe ruptures would not have been included because they would be dominated by failure of other components such as check valves which have failure probabilities several orders of magnitude higher.

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

The proposed TSCR provides an alternative method of leak detection for the required 10-year inservice inspection. It does not result in an operational condition different from that which has already been considered by TS. Therefore, the change does not create the possibility of a new or different kind of accident or malfunction.

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

The alternative method of leak detection has no impact on the consequences of any analyzed accident and does not significantly change the failure probability of equipment which provides protection for the health and safety of the public. Therefore, there is no significant decrease in the margin of safety.

Pursuant to 10 CFR 50.91, the preceding analyses provides a determination that the proposed Technical Specifications change poses no significant hazard as delineated by 10 CFR 50.92.

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

Attorney for licensee: Randolph R. Mahan, South Carolina Electric & Gas

Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment requests: March 30, 2000 (PCN–515).

Description of amendment requests: The amendment application proposes to revise the San Onofre Nuclear Generating Station, Units 2 and 3, Technical Specification (TS) 3.6.6.1, ''Containment Spray and Cooling Systems,'' and the associated Bases. The proposed change would revise the Allowed Outage Time (AOT) for a single inoperable train of the containment spray system from 72 hours to 7 days and revise the combined AOT of 10 days which appears in both Conditions A and C of Limiting Condition for Operation 3.6.6.1 from 10 days to 14 days.

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

(1) Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

This proposed change is a request to revise Technical Specification 3.6.6.1, ''Containment Spray and Cooling Systems'' and the associated Bases. The proposed change revises the Allowed Outage Time (AOT) for a single inoperable train of the Containment Spray System (CSS) from 72 hours to 7 days. The following changes are proposed for the Containment Spray System as described in Technical Specification (TS) 3.6.6.1:

a. The Allowed Outage Time (AOT) for a single train of Containment Spray (Condition A of LCO 3.6.6.1) is extended from 72 hours to 7 days.

b. The Combined AOT of 10 days which appears in both Conditions A and C of LCO 3.6.6.1 is extended from 10 days to 14 days.

c. The Bases of TS 3.6.6.1 are revised to reflect the changes described above.

The Containment Spray System is an Engineered Safety Feature (ESF) system. Inoperable Containment Spray components are not considered to be accident initiators. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

The proposed AOT for the Containment Spray System does impact the ability to mitigate accident sequences. Therefore, to fully evaluate the effects of the proposed CSS AOT extension, Probabilistic Safety Analysis (PSA) methods were utilized. The results of these analyses show no significant increase in core damage frequency. As a result, there would be no significant increase in the consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

This proposed change does not change the design, configuration, or method of operation of the plant.

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

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

Response: No

The proposed change does not affect the limiting conditions for operation or their bases that are used in the deterministic analyses to establish the margin of safety. PSA evaluations were used to evaluate these changes.

Therefore, there will be no significant reduction in a margin of safety as a result of this 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 requests involve no significant hazards consideration.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Stephen Dembek.

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 17, 2000.

Description of amendment request: The proposed changes would modify the voltage setting limits specified in Technical Specification (TS) Table 3.7–4, page 3.7–26, item 7 for the emergency bus degraded voltage, and revise the loss of voltage setpoints from a percentage of nominal bus voltage to an actual bus voltage value. The degraded voltage setting limit is being changed to increase the minimum allowable bus voltage to improve long-term motor performance in the event of operation with bus voltage less than nominal. The emergency bus loss of voltage setting limit is being revised to better address expected relay performance over time (*i.e.,* setting drift). Section 3.6.B, page 3.6–1, of the TS would be changed to revise the required reactor coolant system conditions from the existing wording of ''350 degrees F or 450 psig'' to ''350 degrees F and 450 psig.''

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:

We have reviewed the proposed change against the criteria of 10 CFR 50.92 and have concluded that the change does not pose a significant safety hazards consideration as defined therein. 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.

No increase in the probability of occurrence or consequences of an accident previously evaluated will result from the proposed change in the setting limits for the emergency bus degraded voltage and loss of voltage relay setpoints. The proposed change only affects actuation limits and therefore has no bearing on the probability of an accident. Neither the logic nor the function of the undervoltage protection circuits is being changed, nor is circuit or equipment reliability being reduced. The higher degraded voltage relay setpoint limit will improve motor terminal voltage, and thus promote longer motor life. Changing the setpoint limit for the loss of voltage relays will better characterize the relays' capabilities and facilitate calibration. Further, the performance characteristics of the electrical distribution system and components supplied (motors, etc.) are not being altered, and compliance with GDC–17 [General Design Criterion] is being maintained. The electrical distribution system remains capable of performing its safety function without spurious separation of the emergency buses from offsite power. If offsite power is lost, the capability of the EDG's [emergency diesel generators] to perform their safety function is not altered. Therefore, the probability of an accident previously evaluated is not increased.

The consequences of an accident do not increase since the proposed change implements setting limits that will continue to ensure that adequate voltages will be available for the continuous operation of safety-related equipment required to function to mitigate a design basis accident. The proposed setting limits for the emergency bus degraded voltage and loss of voltage bound the setpoints and initial conditions assumed in the accident analyses and ensure that appropriate protection is maintained.

The editorial change is administrative in nature and consequently does not affect the probability or consequences of an accident in any way.

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

Implementing the proposed Technical Specifications emergency bus degraded voltage and loss of voltage relay setting limits cannot create the possibility of a new or different kind of accident than any accident previously evaluated. Revising the setpoint setting limits does not introduce any new accident precursors, and operation of the electrical distribution system and the undervoltage relaying schemes is unchanged. Raising the setting limit for emergency bus degraded voltage and decreasing the setting limit for emergency bus loss of voltage do not introduce any new accident precursors or modes of operation. The relays will continue to detect undervoltage conditions and transfer safety loads to the emergency diesel generators at a voltage level adequate to ensure proper safety equipment performance and to prevent long-term equipment degradation due to undervoltage conditions. The proposed setting limits include adequate tolerances to calibrate the undervoltage relays while ensuring that emergency bus voltages remain above analytical limits. As noted above, the performance characteristics of the electrical distribution system and the components being supplied are not being altered, and compliance with GDC–17 is being maintained. The proposed Technical Specifications change will ensure that appropriate electrical protection is available as assumed in the safety analysis.

The editorial change is administrative in nature and consequently does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change continues to ensure that adequate voltage is available for safety-related equipment relied upon to respond to a design basis accident. The proposed setting limit for degraded bus voltage is conservative with respect to the existing Technical Specifications and ensures an adequate safety margin is being maintained. Further, the setting limit is maintained low enough to prevent spurious actuations given expected offsite grid voltages. The setting limit for the emergency bus loss of voltage relays is being changed to better characterize the relays' capabilities and to facilitate calibration. While the loss of bus voltage setting limit is being reduced, sustained bus voltage in this range is not credible. Furthermore, there is no safety limit associated with the loss of voltage setting limit.

The proposed change continues to ensure that the setting limits for the emergency bus degraded voltage and loss of voltage relays bound the setpoints and initial conditions assumed in the accident analyses and ensures that appropriate electrical protection is maintained. The editorial change is administrative in nature and consequently does not affect the safety analysis in any way. Consequently, the margin of safety is not being reduced by the proposed Technical Specifications change.

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 Section Chief: Richard L. Emch, Jr.

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.

Carolina Power & Light Company, et al., Docket No. 50–325, Brunswick Steam Electric Plant, Unit 1, Brunswick County, North Carolina

Date of amendment request: April 14, 2000.

Description of amendment request: The proposed amendment would modify Surveillance Requirement 3.1.3.3 to allow partial insertion of control rod 26–47 instead of insertion of one complete notch. This revised acceptance criterion would be limited to the current Unit No. 1 operating cycle, after which the current one-notch requirement will be re-established.

*Date of publication of individual notice in **Federal Register:*** April 21, 2000 (65 FR 21481).

Expiration date of individual notice: May 22, 2000.

GPU Nuclear, Inc., Docket No. 50–320, Three Mile Island Nuclear Station, Unit 2, Middletown, Pennsylvania

Date of amendment request: April 6, 2000.

Brief description of amendment request: The proposed amendment would reflect an administrative name change from GPU Nuclear Corporation to GPU Nuclear, Inc. Furthermore, the proposed license amendment makes an editorial change to better describe TMI–2's use of site physical security, guard training and qualification, and safeguard contingency plans that are maintained by the Three Mile Island Nuclear Station, Unit 1, licensee, AmerGen Energy Company, LLC. In addition, the licensee requests that minor changes (mainly in titles) be made in Section 6.0 of the Technical Specifications to reflect the TMI–2 organizational and administrative controls that will exist following the sale of the Oyster Creek Nuclear Generating Station.

*Date of publication of individual notice in **Federal Register:*** April 21, 2000 (65 FR 21484).

Expiration date of individual notice: May 21, 2000.

Notice of Issance 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).

Baltimore Gas and Electric Company, Docket No. 50–317, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Calvert County, Maryland

Date of application for amendment: February 18, 2000, as supplemented March 3, 2000.

Brief description of amendment: The amendment approved resolution of an issue involving the Societie Alsacienne Construction Mechaniques Del Melhouse (SACM) diesel generator (DG) that constitutes an unreviewed safety question. Specifically, a new failure mode has been identified for DG 1A SACM that is not adequately described in the Updated Final Safety Analysis Report. The manufacturer has indicated that operating the engine in a light load condition may degrade engine performance and ultimately result in engine failure.

Date of issuance: April 20, 2000.

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

Amendment No.: 235.

Renewed Facility Operating License No. DPR–53: Amendment revised the Updated Final Safety Analysis Report.

Date of initial notice in ***Federal Register:*** March 7, 2000 (65 FR 12038).

The March 3, 2000, submittal did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 20, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 27, 1999, as supplemented by letters of February 25, 2000, March 30, 2000, and eMail of March 13, 2000.

Brief description of amendment: This amendment revises the maximum allowable service water temperature permitted by Surveillance Requirement 3.7.8.2 for the ultimate heat sink (UHS) from the currently permitted limit of 95 °F to 97 °F while it restores the original Technical Specifications provisions for required action and completion times of 6/36 hours to be in mode 3/5, respectively, in the event the UHS temperature were to exceed 97 °F.

Date of issuance: April 18, 2000.

Effective date: April 18, 2000.

Amendment No.: 187.

Facility Operating License No. DPR–23: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 23, 2000 (65 FR 9001). The supplements of February 25, March 13, and March 30, 2000, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No

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

Date of application for amendments: May 5, 1999, as supplemented on October 8, 1999.

Brief description of amendments: The amendments resolved an Unreviewed Safety Question (USQ) related to an evaluation of the reactor building ventilation system exhaust plenum masonry walls. The amendments approved the use of different methodology and acceptance criteria for the reassessment of certain masonry walls subjected to transient pressurization loads resulting from a high energy line break. This change to the licensing basis, when evaluated by the licensee in accordance with 10 CFR 50.59, resulted in an USQ that required prior approval by the NRC staff in accordance with the provisions of 10 CFR 50.90.

Date of issuance: April 11, 2000.

Effective date: Immediately, to be implemented during the next scheduled Final Safety Analysis Report update.

Amendment Nos.: 139 and 124.

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Updated Final Safety Analysis Report.

Date of initial notice in ***Federal Register:*** June 16, 1999 (64 FR 32286). The October 8, 1999, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 11, 2000.

No significant hazards consideration comments received: No.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: September 23, 1999.

Brief description of amendment: The amendment relocates items associated with instrumentation for toxic gas monitoring from Technical Specifications to the Updated Final Safety Analysis Report.

Date of issuance: April 20, 2000.

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

Amendment No.: 208

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 1, 1999 (64 FR 67332).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 20, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 22, 1999, as supplemented by letters dated March 20, March 24 (2), March 29, and April 5, 2000.

Brief description of amendment: The amendment authorized revisions to the radiological assessment calculational methodology for the loss-of-coolant accident (LOCA) and the control rod drop accident (CRDA). The amendment request was submitted to address potential unreviewed safety questions resulting from these revisions due to instances of increased dose consequences. Because of outstanding issues involving various assumptions used in these calculational methodologies, the staff is deferring the review of implementing this change on a permanent basis. Subsequently, this amendment is to be effective immediately and remain effective until Cooper Nuclear Station enters mode 4 in preparation for refueling outage 20 (effectively, one operating cycle). Also, the staff has deferred review of the radiological assessment methodology revisions for the fuel handling accident (FHA) and the main steamline break (MSLB) accident. It is anticipated that Nebraska Public Power District (NPPD) will resolve any outstanding issues concerning these calculational methodology revisions in a timely manner in support of a permanent change that is acceptable to the staff.

Date of issuance: April 7, 2000.

Effective date: April 7, 2000, to be implemented within 30 days and remain effective until Cooper Nuclear Station enters mode 4 in preparation for refueling outage 20.

Amendment No.: 183.

Facility Operating License No. DPR–46: The amendment authorizes changes to the licensing basis and changes to the operating license.

Date of initial notice in ***Federal Register:*** January 26, 2000 (65 FR 4280). The March 20 and 24 (2), March 29, and April 5, 2000, letters provided additional clarifying information that was within the scope of the original application and **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 7, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 15, 1999, as supplemented by letters dated February 15 and April 8, 2000.

Brief description of amendment: The technical specification change revises the average power range monitors (APRMs) neutron flux-high (flow-biased) allowable value based on a revised power-to-flow map. The revised power-to-flow map extends the current plant operating domain to above the rated rod line, to within an envelope referred to as the maximum extended load line limit (MELLL) and adds the increased core flow (105 percent) region.

Date of issuance: April 11, 2000.

Effective date: April 11, 2000, to be implemented within 30 days.

Amendment No.: 184.

Facility Operating License No. DPR–46: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 26, 2000 (65 FR 4279). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 11, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: November 19, 1999.

Description of amendment request: The amendment revised Technical Specification (TS) 6.4.3, ''Nuclear Safety Audit Review Committee (NSARC),'' by relocating the specific requirements of this TS to the Quality Assurance Program located in the Updated Final Safety Analysis Report (UFSAR).

Date of issuance: April 11, 2000.

Effective date: As of its date of issuance, and shall be implemented within 90 days.

Amendment No.: 67.

Facility Operating License No. NPF–86: Amendment revised the Technical Specifications and authorized changes to the UFSAR.

Date of initial notice in ***Federal Register:*** January 26, 2000 (65 FR 4281).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 11, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 1, 2000.

Brief description of amendment: The amendment revises limiting conditions for operation (LCO) 3.0.1 and 3.0.2 and adds LCO 3.0.5 to the Technical Specifications (TSs) for Milstone 3. LCO 3.0.5 establishes allowances for restoring equipment to service under administrative controls when the equipment has been removed from service or declared inoperable to comply with actions in the TSs.

Date of issuance: April 17, 2000.

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

Amendment No.: 179.

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

Date of initial notice in ***Federal Register:*** March 1, 2000 (65 FR 11092).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 10, 1999, as supplemented February 25, 2000.

Brief description of amendments: The amendments revise the elevated F-star (EF*) distance for the steam generator tubes specified in Technical Specification 4.12.D.1.(l) following a correction to a minor error in the calculations supporting the current EF* distance.

Date of issuance: April 19, 2000.

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

Amendment Nos.: 149 and 140.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 23, 2000 (65 FR 9010).

The February 25, 2000, supplemental letter provided clarifying information that did not change the staff's initial proposed no significant hazards consideration determination and did not expand it beyond the scope of the original **Federal Register** notice.

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 7, 1999, as supplemented September 27, 1999.

Brief description of amendments: Revised Technical Specifications Section 3/4.4.3 to clarify the action statement concerning inoperative reactor coolant leakage detection systems.

Date of issuance: April 5, 2000.

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

Amendment Nos.: 140 and 103.

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

Date of initial notice in ***Federal Register:*** July 14, 1999 (64 FR 38034).

The September 27, 1999, letter provided clarifying information that did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 5, 2000.

No significant hazards consideration comments received: No.

PECO Energy Company, Docket No. 50–352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of application for amendment: October 14, 1999, as supplemented February 11, 2000.

Brief description of amendment: This amendment revised Technical Specification (TS) Section 2.2, ''Safety

Limits and Limiting Safety Systems Settings,'' and TS Section 3.0/4.0, ''Limiting Conditions for Operation and Surveillance Requirements.''

Date of issuance: April 12, 2000.

Effective date: Effective as of the date of issuance and; Unit 1 shall be implemented during the LGS Unit 1 refueling outage scheduled to begin March 29, 2000.

Amendment No.: 141.

Facility Operating License No. NPF–39. This amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 1, 1999 (64 FR 67337).

The February 11, 2000, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 12, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 26, 1998, as supplemented October 14, 1999.

Brief description of amendment: The amendment changes the Technical Specifications (TSs) by changing the value of the allowable containment leakage rate to 1.5 percent per day and correcting conflicting information in TS Section 4.6.C, ''Coolant Chemistry.''

Date of issuance: April 14, 2000.

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

Amendment No.: 261.

Facility Operating License No. DPR–59: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 22, 1998 (63 FR 19977).

The October 14, 1999, supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 30, 1999.

Brief description of amendment: The amendment revises Technical Specification 5.5.10, ''Ventilation Filter Testing Program'' to meet the actions requested by Generic Letter 99–02.

Date of issuance: April 12, 2000.

Effective date: April 12, 2000.

Amendment No.: 77.

Facility Operating License No. DPR–18: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 26, 2000 (65 FR 4290)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 12, 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: February 26, 1999.

Brief description of amendments: These amendments revised the Technical Specifications (TSs) to eliminate inconsistencies and redundancies in Section 3.8.1.1, action statements involving inoperable offsite AC circuits and combinations of inoperable offsite power supplies and emergency diesel generators.

Date of issuance: April 14, 2000.

Effective date: April 14, 2000.

Amendment Nos.: 255 and 246.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revised the TSs.

Date of initial notice in ***Federal Register:*** March 24, 1999 (64 FR 14287).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 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 June 2, 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).

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

Date of application for amendments: March 29, 2000 (TS–402).

Brief Description of amendments: The amendments revised the Technical Specifications (TS) requirements applicable to opening of secondary containment access doors.

Date of issuance: April 21, 2000.

Effective date: April 21, 2000.

Amendment Nos.: 238, 264, and 224.

Facility Operating License No. DPR–33, DPR–52 and DPR–68: Amendments revise the TS.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes (65 FR 18141 dated April 6, 2000). The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided for an opportunity to request a hearing by April 20, 2000, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of NSHC are contained in a Safety Evaluation dated April 21, 2000.

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

NRC Section Chief: Richard P. Correia.

Dated at Rockville, Maryland, this 26th day of April 2000.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–10743 Filed 5–2–00; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

Privacy Act of 1974; Amendment to a System of Records

AGENCY: Office of Personnel Management.

ACTION: Technical amendment of existing routine use.

SUMMARY: This notice serves as a technical amendment to an existing routine use contained in OPM's CENTRAL–1 system of records.

DATES: The change will be effected without further notice on June 12, 2000 unless comments are received that would result in a contrary determination.

ADDRESSES: Send written comments to the Office of Personnel Management, ATTN: Mary Beth Smith-Toomey, Office of the Chief Information Officer, 1900 E Street NW., Room 5415, Washington, DC 20415–7900.

FOR FURTHER INFORMATION CONTACT: Mary Beth Smith-Toomey, (202) 606–8358.

SUPPLEMENTARY INFORMATION: In OPM's CENTRAL–1 system of records, routine use(s) has been amended to move ''requesting'' in front of the word ''States'' to clarify that OPM can disclose information to Federal agencies regardless of whether they specifically requested the information.

(s) To disclose information contained in the Retirement Annuity Master File; including the name, Social Security Number, date of birth, sex, OPM's claim number, health benefit enrollment code, retirement date, retirement code (type of retirement), annuity rate, pay status of case, correspondence address, and ZIP code, of all Federal retirees and their survivors to Federal agencies and requesting States to help eliminate fraud and abuse in the benefit programs administered by the Federal agencies and States (and those States to local governments) and to collect debts and overpayments owed to the Federal Government, and to State governments and their components.

Office of Personnel Management.

Janice R. Lachance,

Director.

[FR Doc. 00–10989 Filed 5–2–00; 8:45 am]

BILLING CODE 6325–01–P

SECURITIES AND EXCHANGE COMMISSION

Proposed Collection; Comment Request

Upon Written Request, Copies Available From: Securities and Exchange Commission, Office of Filings and Information Services, Washington, DC 20549.

Extension:

Rule 17f–1(b)—SEC File No. 270–28—OMB Control No. 3235–0032

Rule 17f–1(c) and Form X–17F–1A—SEC File No. 270–29—OMB Control No. 3235–0037

Rule 17h–1T and 17h–2T—SEC File No. 270–359—OMB Control No. 3235–0410

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*) the Securities and Exchange Commission (''Commission'') is soliciting comments on the collections of information summarized below. The Commission plans to submit these existing collections of information to the Office of Management and Budget for extension and approval.

Rule 17f–1(b) requires approximately 1,150 entities in the securities industry to register in the Lost and Stolen Securities Program. Registration fulfills a statutory requirement that entities report and inquire about missing, lost, counterfeit, or stolen securities. Registration also allows entities in the securities industry to gain access to a confidential database that stores information for the program.

It is estimated that 1,150 entities will register in the Lost and Stolen Securities Program annually. It is also estimated that each respondent will register one time. The staff estimates that the average number of hours necessary to comply with the Rule 17f–1(b) is one-half hour. The total burden in 575 hours annually for respondents, based upon past submissions. The average cost per hour is approximately $50. Therefore, the total cost of compliance for respondents is $28,750.

Rule 17f–1(b) is a reporting rule and does not specify a retention period. The rule requires a one-time registration for reporting institutions. Registering under Rule 17f–1(b) is mandatory to obtain the benefit of a central database that stores information about missing, lost, counterfeit, or stolen securities for the Lost and Stolen Securities Program. Reporting institutions required to registered under Rule 17f–1(b) will not be kept confidential, however, the Lost and Stolen Securities Program database will be kept confidential.

Rule 17f–1(c) and Form X–17F–1A requires approximately 23,000 entities in the securities industry to report lost, stolen, missing, or counterfeit securities to a central database. Form X–17F–1A facilitates the accurate reporting and precise and immediate data entry into the central database. Reporting to the central database fulfills a statutory requirement that reporting institutions report and inquire about missing, lost, counterfeit, or stolen securities. Reporting to the central database also allows reporting institutions to gain access to the database that stores information for the Lost and Stolen Securities Program.

It is estimated that 23,000 reporting institutions will report that securities are either missing, lost, counterfeit, or stolen annually. It is also estimated that each reporting institution will submit this report 56 times each year. The staff estimates that the average amount of time necessary to comply with Rule 17f–1(c) and Form X–17F–1A is five minutes. The total burden is 107,333 hours annually for respondents, based upon past submissions. The average cost per hour is approximately $50. Therefore, the total cost of compliance for respondent is $5,366,666.

Rule 17f–1(c) is a reporting rule and does not specify a retention period. The rule requires an incident-based reporting requirement by the reporting institutions when securities are discovered missing, lost, counterfeit, or stolen. Registering under Rule 17f–1(c) is mandatory to obtain the benefit of a central database that stores information about missing, lost, counterfeit, or stolen securities for the Lost and Stolen Securities Program. Reporting institutions required to register under Rule 17f–1(c) will not be kept confidential, however, the Lost and Stolen Securities Program database will be kept confidential.

Rules 17h–1T requires a broker-dealer to maintain and preserve records and other information concerning certain entities that are associated with the