Dated: April 30, 1999.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

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NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from April 10, 1999, through April 23, 1999. The last biweekly notice was published on April 21, 1999 (64 FR 19554).

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

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

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

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

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

By June 4, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW, Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

Date of amendment request: April 12, 1999.

Description of amendment request: The proposed one-time technical specification (TS) change, effective through September 30, 1999, provides a Required Action and Completion Time for the Ultimate Heat Sink (UHS) in the event that service water temperature exceeds the current 95°F surveillance limit. It involves an allowance to continue operation for a period of 8 hours with the UHS at a temperature greater than the temperature limits provided in TS Limiting Condition of Operation 3.7.8, ''Ultimate Heat Sink (UHS)'' and provides an upper UHS temperature limit beyond which plant shutdown is required.

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

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

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

The proposed change does not involve any physical alteration of plant systems, structures or components. The proposed change allows plant operation for a short period of time when the service water temperature exceeds 95°F, requires an hourly surveillance when service water temperature exceeds 95°F, provides an upper UHS temperature limit beyond which a plant shutdown is required, and specifies an expiration date beyond which the current requirements are restored. If the service water temperature is restored within the allowed time, a plant shutdown is not required. This minimizes plant transients, which reduces the probability of a reactor trip and the resulting challenges to mitigating systems. A service water temperature of up to 99°F does not increase the failure rate of systems, structures or components because the systems, structures, and components are designed for higher temperatures than at which they operate.

The Service Water (SW) System temperature is not assumed to be an initiating condition of any accident evaluated in the safety analysis report. Therefore, the allowance of a limited time for service water temperature to be in excess of 95°F does not involve an increase in the probability of an accident previously evaluated in the safety analysis report (SAR). The SW System supports operability of safety related systems used to mitigate the consequences of an accident. The service water temperature is not expected to increase significantly beyond 95°F due to the limited time allowed by the proposed change in conjunction with the generally slow rate of temperature increase experienced from thermal changes in Lake Robinson. The capability of components to perform their safety related function is not affected up to a service water temperature of 99°F with the exception of the Containment Air Recirculation Fan Coolers. The heat removal capacity of the Containment Air Recirculation Fan Coolers is not expected to be significantly reduced by a small increase in service water temperature. If heat removal is not significantly reduced, containment pressure and leakage will not be significantly increased, and the doses from containment leakage will not be significantly increased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the SAR.

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

The proposed change does not involve any physical alteration of plant systems, structures or components. A service water temperature of up to 99°F does not introduce new failure mechanisms of systems, structures or components not already considered in the SAR because the systems, structures, and components are designed for higher temperatures than at which they operate. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

The proposed change will allow a small increase in service water temperature above the design basis limit for the SW System and delay by 8 hours the requirement to shutdown the plant when the service water system design limit is exceeded. There are design margins associated with systems, structures and components that are cooled by the service water system that are affected. The capability of components to perform their safety related function is not affected up to a service water temperature [of] 99°F with the exception of the Containment Air Recirculation Fan Coolers. The Containment Air Recirculation Fan Coolers remove heat from containment to mitigate containment pressure and temperature following a MSLB [main steamline break] inside containment or a Large Break LOCA [loss-of-coolant accident] inside containment. An increase in service water temperature in excess of the design limit due to hot weather conditions is expected to be small due to the limited time allowed by the proposed change in conjunction with the generally slow rate of temperature increase experienced from thermal changes in Lake Robinson. Therefore, the effect on the Containment Air

Recirculation Fan Coolers' heat removal capacity and the resulting containment pressure and temperature is expected to be small. Therefore, there is no significant reduction in margin of safety associated with 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 request involves no significant hazards consideration.

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

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

NRC Section Chief: Sheri R. Peterson.

Commonwealth Edison Company, Docket No. 50–254, Quad Cities Nuclear Power Station, Units 1, Rock Island County, Illinois

Date of amendment request: March 30, 1999.

Description of amendment request: The amendment would revise the Quad Cities Nuclear Power Station, Unit 1 Technical Specifications (TS) by changing the Surveillance Requirements (SR) 4.6.E.2 to allow a one-time extension of the 18-month requirement to pressure set test or replace one half of the Main Steam Safety Valves (MSSVs) to an interval of 24 months.

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

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

The proposed changes request a one-time change to the surveillance requirement for the MSSVs and Target Rock S/RV [Safety Relief Valve]. The surveillance interval between MSSVs and Target Rock S/RV testing is not a precursor assumed in any previously analyzed accident. Therefore, the probability of a previously evaluated accident has not been increased.

The proposed extension is consistent with the ASME Code requirement to test 20% of the sample population every 24 months with all of the valves in the sample group being tested every 60 months. The proposed changes are also consistent with NUREG 1433, Revision 1, and do not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. Operating experience and excellent materiel condition of the MSSVs and Target Rock S/RV support the expectation that they will continue to perform their intended function. Therefore, the consequences of a previously evaluated accident have not been increased.

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

No new equipment is required, nor will the MSSVs and Target Rock S/RV be operated in a different manner during the period of the extended surveillance interval. The proposed changes are consistent with NUREG 1433, Revision 1, requirements for safety valve surveillance intervals as well as the ASME Code requirements for testing safety valves. Operating experience and superior materiel condition of the MSSVs and Target Rock S/RV support the expectation that they will continue to perform their intended function. Therefore, the possibility of a new or different accident has not been increased.

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

The proposed amendment represents an extension to the current TS SRs that would otherwise be provided generically by the ASME Code. The proposed changes are also consistent with NUREG–1433, Revision 1, and do not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. The proposed changes have been evaluated and found to be acceptable for use at Quad Cities Nuclear Power Station based on system safety analysis requirements and operational performance. The MSSVs and Target Rock S/RV provisions continue to be adequately maintained during plant operation. The proposed changes to the MSSVs and Target Rock S/RV surveillance interval do not significantly reduce existing plant safety margins since excellent materiel condition and acceptable surveillance test results support the expectation that no significant degradation will occur over the extended interval.

The proposed changes are based on NRC accepted provisions at other operating plants that are applicable at Quad Cities Nuclear Power Station and maintain necessary levels of system or component reliability.

The proposed amendment for Quad Cities Nuclear Power Station will not reduce the availability of systems required to mitigate accident conditions.

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

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

Local Public Document Room location: Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: March 30, 1999.

Description of amendment request: This amendment request proposes to change the Technical Specifications (TSs) to allow an alternate methodology for quantifying Reactor Coolant System (RCS) leakage when the normal RCS leakage detection system is inoperable.

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

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

The current Technical Specifications require a periodic measurement of RCS leakage. The normal method for quantifying RCS leakage is to use the DWFDS [Drywell Floor Drain Sump] and DWEDS [Drywell Equipment Drain Sump] flow totalizers. The proposed TS change would allow an alternate method for quantifying RCS leakage when a flow totalizer is not available. The proposed change has no impact on the frequency for monitoring RCS leakage and would only be used for a maximum of 30 days while the normal leakage monitoring system is being restored to an operable condition. The alternate methodology for quantifying leakage has a measurement sensitivity that is consistent with the normal method. The proposed change does not impact any system structure or component used to mitigate the consequences of an accident and there will be no change in the types or significant increase in the amounts of any effluents released offsite.

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

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

The proposed change involves no physical modifications to any system, structure or component used to mitigate the consequences of an accident. The operation of the DWEDS and DWFDS are not being altered in any way that could affect their ability to function during an accident condition.

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

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

The current TS's require a periodic measurement of RCS leakage. The normal method for quantifying RCS leakage is to use the DWFDS and DWEDS flow totalizers. The proposed technical specifications change would allow an alternate method for quantifying RCS leakage when a flow totalizer is inoperable. The proposed change has no impact on the frequency for monitoring RCS leakage and would only be used for a maximum of 30-days while the normal leakage monitoring system is being restored to an operable condition. The proposed alternate methodology for quantifying leakage has a measurement sensitivity that is consistent with the normal method.

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

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

Local Public Document Room location: Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: March 30, 1999.

Description of amendment request: This amendment request proposes to revise license conditions in each of the respective Operating Licenses to delete those license conditions that no longer apply, make an editorial change in the Unit 1 license, and provide clarifying information regarding the license condition concerning equalizer valve restrictions.

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

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

The initial conditions and methodologies used in the accident analyses remain unchanged. The proposed changes do not change or alter the design assumptions for the systems or components used to mitigate the consequences of an accident. Therefore, accident analyses results are not impacted.

The proposed changes delete various license conditions that have been completed, make editorial changes, and provide clarifying information. The changes are administrative. No physical or operational changes to the facility will result from the proposed changes.

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

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

The proposed changes do not affect the design or operation of any system, structure, or component in the plant. The safety functions of the related structures, systems, or components are not changed in any manner, nor is the reliability of any structures , systems, or component reduced. The changes do not affect the manner by which the facility is operated and do not change any facility design feature, structure, system, or component. No new or different type of equipment will be installed.

The proposed changes delete various license conditions that have been completed, make editorial changes, and provide clarifying information. The changes are administrative. No physical or operational changes to the facility will result from the proposed changes.

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

Does the change involve a significant reduction in the margin of safety for the following reasons:

The proposed changes are administrative in nature and have no impact on the margin of safety of any Technical Specification. There is no impact on safety limits or limiting safety system settings. The changes do not affect any plant safety parameters or setpoints. The proposed changes delete various license conditions that have been completed, make editorial changes, and provide clarifying information. No physical or operational changes to the facility will result from the proposed changes.

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

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

Local Public Document Room location: Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: March 25, 1999.

Description of amendment request: The amendments would revise the Technical Specifications (TS) to redefine the ‘‘trip setpoint’’ in a number of locations as the ‘‘nominal trip setpoint.’’ The current definition results in upper-or lower-bound numerical values not to be exceeded for setpoints. This proposed new definition would permit the setpoints to be set within a tolerance range around the number specified in various tables. The TS locations affected are: Table 3.3.1–1, ‘‘Reactor Trip System Instrumentation;’’ Table 3.3.2–1, ‘‘Engineered Safety Feature Actuation Instrumentation;’’ Surveillance Requirement 3.3.5.2; Table 3.3.6–1, ‘‘Containment Purge and Exhaust Isolation Instrumentation;’’ and Limiting Condition of Operation (LCO) 3.4.12. Sections of the associated TS Bases document would also be revised to reflect the TS changes.

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?

No. The proposed changes are consistent with the current licensing basis for Catawba Nuclear Station, the setpoint methodology used to develop the Trip Setpoints, the Catawba Safety Analyses, and current station calibration procedures and practices. The Reactor Trip System and Engineered Safety Features Actuation System are not accident initiating systems; they are accident mitigating systems. Therefore, these proposed changes will have no impact on any accident probabilities. Accident consequences will not be affected, as no changes are being made to the plant which will involve a reduction in reliability of these systems. Consequently, any previous evaluations associated with accidents will not be affected by these changes.

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

No. The proposed changes are consistent with the current licensing basis for Catawba Nuclear Station, the setpoint methodology used to develop the Trip Setpoints, the Catawba Safety Analyses, and current station calibration procedures and practices. No changes are being made to actual plant hardware which will result in any new accident causal mechanisms. Also, no changes are being made to the way in which the plant is being operated. Therefore, no new accident causal mechanisms will be generated. Consequently, plant accident analyses will not be affected by these changes.

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

No. The proposed changes are consistent with the current licensing basis for Catawba Nuclear Station, the setpoint methodology used to develop the Trip Setpoints, the Catawba Safety Analyses, and current station calibration procedures and practices. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following accident conditions. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these barriers will not be degraded by the proposed changes. Consequently, plant safety analyses will not be affected by these changes.

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

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

Attorney for licensee: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: April 9, 1999.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TSs) to add Limiting Condition for Operation (LCO) 3.0.6 and its associated bases. This change would allow equipment that has been removed from service or declared inoperable in compliance with the TS Action statement to be returned to service under administrative controls solely to perform testing required to demonstrate its operability or the operability of other equipment. The proposed change is consistent with TS 3.0.5 as discussed in NUREG–1432, Revision 1, ''Standard Technical Specifications for Combustion Engineering Plants.'' TS 3.0.2 would also be modified to reflect that TS 3.0.6 is an exception to TS 3.0.2.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. 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 change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change would allow an orderly return to service of inoperable equipment. This change does not alter the functional characteristics of any plant component and does not allow any new modes of operation of any component. The accident mitigation features of the plant are not affected by the proposed amendment request. Therefore, this proposed amendment would not result in a significant change in the types or significant increase in the amounts of any effluents that may be released off site. No modifications to the plant have been proposed due to this amendment request. The proposed change would permit equipment removed from service to comply with required actions to be returned to service under administrative controls to verify the operability of the equipment being returned to service or of other related equipment. Although returning inoperable equipment to service for testing may temporarily compromise single failure criteria, administrative controls will ensure the time involved will be limited to only that required to demonstrate component or system operability. This LCO provides an acceptable method of restoring equipment to service for the sole purpose of demonstrating its operability or the operability of other related equipment. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

No modifications were made to the plant due to this amendment request. The proposed change does not alter the functional characteristics of any plant component and does not allow any new modes of operation for any component. This proposed amendment would facilitate the testing of equipment in its design configuration to demonstrate operability. The use of TS 3.0.6 would be limited to the time absolutely necessary to perform the test. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The addition of TS 3.0.6 is considered necessary to establish an allowance that is not formally recognized in the current TSs. Without this allowance, situations can arise in which certain components could not be restored to operable status without requiring a plant shutdown. It is not the intent that the TSs preclude the return to service of a component to confirm its operability. This allowance is deemed to represent a more stable, safe operation than requiring a plant shutdown to complete the restoration and confirmatory testing. The time period during which the equipment is returned to service in conflict with the requirements of the TS Action statement is limited to the time absolutely necessary to perform the indicated surveillance requirement. TS 3.0.6 does not provide time to perform any other preventive or corrective maintenance. The period of time during which the equipment is returned to service will be limited by administrative controls and is considered very small. Therefore, the probability of an accident during that time period is also very small and is considered to be insignificant. Thus, it can be concluded that the proposed change does not affect the current margin of safety.

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

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

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

NRC Section Chief: Robert A. Gramm.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of amendment request: March 9, 1999.

Description of amendment request: The proposed change would modify the Technical Specifications to increase the inservice inspection interval, and reduce the scope of volumetric and surface examinations for the reactor coolant pump flywheels.

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

The Nuclear Regulatory Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazard exists due to a proposed amendment to an Operating License for a facility. A proposed amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed changes would: (1) Not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) Not create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Not involve a significant reduction in a margin of safety. The Davis-Besse Nuclear Power Station has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, (DBNPS) Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no accident initiators, conditions, or assumptions are affected by the proposed changes to Technical Specification Surveillance Requirement 4.4.10.1.a in the frequency and scope of volumetric and surface examinations for the Reactor Coolant Pump (RCP) motor flywheels.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because changes in the frequency and scope of volumetric and surface examinations for the RCP motor flywheels will not affect any previously evaluated accidents. Accidents associated with failure of the flywheel were not evaluated in the DBNPS Updated Safety Analysis Report (USAR). The design, fabrication, and testing of flywheels in accordance with the guidance found in NRC Regulatory Guide 1.14, ''Reactor Coolant Pump Flywheel Integrity,'' Revision 1, August 1975, minimizes the potential for flywheel failure. The proposed changes have been demonstrated to maintain conservative testing requirements for the flywheels.

2. Not create the possibility of a new or different kind of accident from any previously evaluated because changes in the frequency and scope of volumetric and surface examinations for the RCP motor flywheels will not affect the reliability of RCP motor flywheels. No new failure mode is introduced since the proposed changes do not involve a modification or change in operation of any plant systems, structures, or components.

3. Not involve a significant reduction in the margin of safety. As shown in Westinghouse Topical Report WCAP–14535A, ''Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination,'' November 1996, RCP motor flywheels have been inspected for twenty years without any service induced flaws being identified. Additionally, the analyses demonstrated that the flywheels are manufactured from high quality steel, have a high fracture toughness, and have a very high flaw tolerance. The topical report indicates that the flywheels could be operated for forty years without inspection, and there would be no significant increase in the probability of failure of the flywheels. However, inspections are proposed to continue at a frequency of once every ten years as a conservative measure. Thus, the margin of safety is not reduced significantly by the proposed change in inspection frequency.

Based on the above, the Davis-Besse Nuclear Power Station has determined that the License Amendment Request does not involve a significant hazards consideration. As this License Amendment Request concerns a proposed change to the Technical Specifications that must be reviewed by the Nuclear Regulatory Commission, this License Amendment Request does not constitute an unreviewed safety question.

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

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, OH 43606.

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: March 31, 1999.

Description of amendment request: The proposed change would modify Cooper Nuclear Station's technical specification administrative controls for unit staff qualifications for the shift supervisor, senior operator, licensed operator, shift technical advisor, and radiological manager.

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

The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change provides enhancement to the current requirements and clarifies the qualifications and training requirements for the shift supervisor, senior operator, licensed operator, shift technical advisor, and Radiological Manager. This provides additional assurance that these personnel are properly trained and qualified for their positions; therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change will not create the possibility of a new or different kind of accident than evaluated in the Updated Safety Analysis Report (USAR). The proposed change provides enhancement to the current requirements and clarifies the qualifications and training requirements for the shift supervisor, senior operator, licensed operator, shift technical advisor, and Radiological Manager. The revised administrative controls for unit staff qualifications are an enhancement to the current requirements; therefore, the proposed change does not create the possibility of a new or different kind of accident.

The proposed change will not create a significant reduction in the margin of safety. The proposed change provides enhancement to the current requirements and clarifies the qualifications and training requirements for the shift supervisor, senior operator, licensed operator, shift technical advisor, and Radiological Manager. This provides additional assurance that these personnel are properly trained and qualified for their positions; therefore, the proposed change will not create a significant reduction in the margin of safety.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: March 31, 1999.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Table 3.6.1.2–1, ''Allowable Leak Rates through Valves in Potential Bypass Leakage Paths,'' by adding two relief valves, with associated leak rate criteria, to be installed on the drywell equipment drain line and drywell floor drain line during the refueling outage in the spring of 2000. Specifically:

(i) For the drywell equipment drain line, the reference to the inboard isolation valve (2DER*MOV119) would be replaced with a reference to the isolation valve and its associated relief valve (2DER*MOV119 and 2DER*RV344);

(ii) For the drywell floor drain line, the reference to the inboard isolation valve (2DFR*MOV121) would be replaced with a reference to the isolation valve and its associated relief valve (2DFR*MOV121 and 2DFR*RV228); and

(iii) A footnote for both above changes would be added to state, ''For valves 2DER*MOV 119 and 2DER*RV344, and likewise for valves 2DFR*MOV121 and 2DFR*RV228, this limit shall be the combined allowable leak rate and not the per valve allowable leak rate.''

The two relief valves would be installed to protect the drain line penetrations against overpressure, consistent with Generic Letter 96–06, ''Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions.'' The allowable leak rates currently specified in TS Table 3.6.1.2–1 for the drywell equipment and drywell floor drain line penetrations will not be increased as a result of the hardware modifications or proposed TS 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:

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

The proposed amendment will add one relief valve on the drywell equipment drain line (penetration 2DER*Z40) and one relief valve on the drywell floor drain line (penetration 2DFR*Z39). These valves will be installed on piping between the inboard containment isolation valve and the primary containment wall. These drain lines represent potential bypass leakage paths from the primary containment to the environment and are subject to maximum allowable isolation valve leak rates, as specified in Table 3.6.1.2–1 of the Technical Specifications (TS). The purpose of adding relief valves is to protect the piping between the inboard and outboard isolation valves against thermally induced overpressure under postulated accident conditions when both isolation valves close, and the fluid trapped between them may heat up and expand. The new relief valves and piping will not cause any existing plant design, operating, or testing limits to be exceeded. The relief valve installations will meet standards and specifications currently applicable to the penetrations being modified. The relief valve configuration, set pressure, and testing meet applicable NRC guidance. No different precursors or new accident initiators are introduced as the result of the proposed modification. Therefore, this proposed amendment does not involve a significant increase in the probability of an accident previously evaluated.

The existing requirements relating to allowable bypass leakage for the two penetrations affected by this modification, will not be changed. No new bypass leakage paths to the environment will be created and no new failure modes will be introduced. Should the relief valves open and fail to close, the effectiveness of the containment and other fission product barriers will not be compromised. As a result, accident dose rates will remain unchanged and within the limits of 10 CFR 50, Appendix A, General Design Criterion 19, and 10 CFR 100. None of the accident assumptions described in Section 6.2, titled ''Containment Systems'' and Chapter 15, titled ''Accident Analysis,'' of the NMP2 Updated Safety Analysis Report (USAR) is adversely affected by the proposed modifications. Therefore, this proposed amendment does not involve a significant increase in the consequences of an accident previously evaluated.

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

The isolation valves associated with penetrations 2DER*Z40 (drywell equipment drain line) and 2DFR*Z39 (drywell floor drain line) perform an accident mitigation function by isolating the containment during and after certain postulated accidents. The addition of relief valves between the inboard and outboard isolation valves will enhance the capability of the existing isolation valves to perform their function without the risk of failure due to piping overpressurization. Consistent with the guidance in Generic Letter 96–06, the consequences of a stuck-open relief valve malfunction have been evaluated and are acceptable. Should the relief valve fail to close after opening, the existing outboard isolation valve will perform its function to isolate the containment. Therefore, operation of NMP2 in accordance with this proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The proposed installation of the relief valves will not adversely affect primary containment integrity, the maximum allowable leak rates for the affected penetrations, any other fission product barriers, or any plant safety/operational limits. The relief valves will assure that the associated isolation valves do not fail as the result of piping overpressure during and after postulated accidents, which will preserve the radiological margin of safety. Therefore, operation of NMP2 in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

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

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

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

NRC Section Chief: S. Singh Bajwa

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

Date of amendment request: March 2, 1999.

Description of amendment request: The proposed amendment would require two service water (SW) pumps and their associated strainers to be operable to declare a service water system (SWS) loop operable. The proposed amendment would also (1) modify the existing action statement to take into account one or more service water pump(s) or strainers being inoperable and (2) make changes to the appropriate Bases section.

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:

NNECO has reviewed the proposed revision in accordance with 10 CFR50.92 and has concluded that the revision does not involve any Significant Hazards Considerations (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not satisfied. The proposed Technical Specification revision does not involve an SHC because the revision would not:

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

The proposed TS [Technical Specification] change adds an additional AOT [allowed outage time] for one of four of the service water pumps/strainers in the SWS. The capabilities of the SWS were evaluated in order to ensure that a significant increase in the probability or consequences of the following previously evaluated accidents, LOP [loss of power], LOCA [loss-of-coolant accident] with concurrent LOP and secondary side piping break inside containment, are precluded by SWS mitigative functions. As the above DBA's [design basis accidents] are not caused by the failure of the SWS to operate, the SWS can not affect the probability of these accidents to occur.

Since both pumps/strainers in each loop are covered by the ACTION statement in the TS when inoperable (due to failure or maintenance), and the proposed ACTION statement for two inoperable service water pumps in a single loop is consistent with the current ACTION statement, there is no impact on the capability to maintain core decay heat removal following a DBA. Further, the revised TS will improve availability of the SWS. The LCO [limiting condition for operation] and ACTION statements help ensure that the SWS, including pumps/strainers, are kept in a condition which allows it to perform all its design functions including providing core decay heat removal and the SFP [spent fuel pool] cooling. As such, there is no affect on the consequences of previously evaluated accidents.

Thus, it is concluded that the proposed revision does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The SWS is used to remove heat from the reactor plant auxiliary systems and other systems. Only one of four pumps is required to be operating during normal plant conditions. In addition, only one 100% capacity pump is required to provide the necessary flow to mitigate the consequences of a DBA. This change continues to require two pumps/strainers per loop to be operable and imposes strict controls on the AOT for the SWS pumps/strainers via the imposition of the LCO controls on the SWS. This assures that four service water pumps/strainers will always be available or the plant will be in an ACTION STATEMENT. The SWS is used to mitigate the consequences of an accident and will not cause an accident.

Thus, this proposed revision 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.

This change will have no impact on the performance of any safety related system covered by the TS. This change explicitly defines the number of pumps/strainers required for the SWS to be considered OPERABLE and the ACTION required which specifies the AOT for inoperable components. The required flow rate for accident mitigation continues to be available to all ECCS [emergency core cooling system] components and their support systems. As such, this change does not increase the peak clad temperature for a DBA–LOCA.

The proposed Technical Specification change adds an additional AOT for one of four of the service water pumps/strainers in the SWS. Two service water pumps/strainers are required to perform the design function of the SWS; one pump to mitigate the DBA and the other to reduce the potential of the SFP boiling which could occur if a service water pump is unavailable for SFP cooling after a design basis LOCA.

The existing TS Bases states that ‘‘The OPERABILITY of the Service Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.’’

Since this change continues to control the availability of the SW pumps by placing the system in an ACTION statement with one loop out of service, then the change will continue to comply with the existing BASES requirements. Thus it is concluded that the proposed revision does not involve a significant reduction in the margin of safety.

In conclusion, based on the information provided, it is determined that the proposed revision does not involve a SHC.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of application for amendments: December 24, 1998.

Description of amendment request: Revises the setpoints and limits of allowable values for loss of power (LOP) instrumentation for 4kV emergency busses.

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

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

The LOP instrumentation provides safety-related electrical equipment protection. No new equipment is added to the plant as a result of the proposed changes. Separation of the 4kV emergency buses from the grid is the only potential transient that previously existed based on operation of these relays. Based on the revised Voltage Regulation Study, which incorporates the effects of system improvements and additional conservatisms, there is no significant increase in the probability of this separation. The relay time delay settings are such that the relays will detect and respond to an actual sustained degradation of voltage, but will not actuate in response to normal operational voltage fluctuations. No accident initiators will be impacted by the proposed setpoint changes. All safety systems will be able to perform their safety functions. Accident mitigation is achieved by these relays by ensuring adequate voltage is maintained throughout the Class 1E electrical distribution system.

The existing allowable values and the proposed allowable values for Functions 2, 3, 4, and 5 have been analyzed and both values are acceptable for operation. During implementation of modification 96–01511 (changing of the relay setpoints), the 4kV buses could be in one of the three configurations: (a) Both sources have relays set at the existing setpoints, (b) one set of source relays with the existing old setpoints and the other set with the proposed revised setpoints, or (c) both sources have relays set at the proposed revised setpoints. Each of these configurations is acceptable because the existing and proposed values satisfy the design limits established within the setpoint calculation and the Voltage Regulation Study.

For Function[s] 4 and 5, the present TS has separate entries in Table 3.3.8.1–1, for the internal and external time delay. This proposed change will combine these internal and external time delays for simplicity. The aggregate time delay is the important parameter and it is the only time delay that is analyzed. The internal time delay minimizes the relay contact wear and reduces the number of external time delay relay actuations due to transient voltage dips. The internal time delay provides no other output functions. Therefore, there will be no impact on the Class 1E power distribution system to perform its intended design function.

Therefore, the proposed changes described above, or operation while modification 96–01511 is being implemented, does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed LOP instrumentation setpoint changes will not result in any new accidents or operational transients. Separation of the 4kV emergency buses from the grid is the only potential transient that previously existed based on operation of these relays. Based on the revised Voltage Regulation Study, which incorporates the effects of system improvements and additional conservatisms, there is no significant increase in the probability of this separation, and the proposed setpoint changes would not create the possibility of a new or different kind of accident from any previously evaluated. The relay time delay settings are such that the relays will detect and respond to an actual sustained degradation of voltage, but will not actuate in response to normal operational voltage fluctuations. The proposed setpoint changes for these relays and the proposed combining of the internal and external time delays will not become initiators of different types of accidents or transients. Additionally, since the existing and proposed allowable values for the LOP instrumentation functions are within the band established by the Voltage Regulation Study, both values are acceptable for operation during the implementation of modification 96–01511. Therefore, the possibility of a new or different kind of accident than previously evaluated is not created.

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

All LOP instrumentation functions will continue to be carried out. The proposed setpoint and allowable value changes have been evaluated within the Voltage Regulation Study and the Plant Electrical Load Study. The relay setpoints have been established using IISCP setpoint methodology. The setpoint determination accounts for relay accuracy, potential transformer accuracy, measurement and test equipment accuracy, and margin above the design limit established within the Voltage Regulation Study. The proposed setpoint changes for these relays and the proposed combining of the internal and external time delays will not involve a significant reduction in a margin of safety. Additionally, since the existing and proposed allowable values for the LOP instrumentation functions are within the band established by the Voltage Regulation Study, both values are acceptable for operation during the implementation of modification 96–01511. Therefore, having both values during the implementation of modification 96–01511 does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Elinor G. Adensam

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

Date of application for amendments: February 12, 1999.

Description of amendment request: Administrative changes to correct typographic errors in Technical Specifications (TS) introduced in previous amendments.

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

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

The proposed changes correct typographical errors and are administrative only and do not impact the operation of the facility. In each case, the action of the intended TS requirements were satisfactorily completed when the change was implemented. These corrections are administrative only and have no effect on any previously evaluated accident scenario. The changes will not alter the operation of equipment assumed to be available for the mitigation of accidents or transients, nor will they alter the operation of equipment important to safety previously evaluated.

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

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

The proposed changes correct typographical errors and are administrative only and will not involve any physical changes to the plant SSCs [systems, structures, or components]. In each case, the action of the intended TS requirements were satisfactorily completed when the change was implemented. These corrections are administrative only and have no effect on any previously evaluated accident scenario. The proposed changes do not allow operation in any mode that is not already evaluated. The changes will not alter the operation of equipment important to safety previously evaluated.

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

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

The proposed changes correct typographical errors and are administrative only and will not affect the manner in which the facility is operated, or change equipment or features which affect the operational characteristics of the facility. The proposed changes have no impact on any safety analysis assumptions or margins of safety.

Therefore, these 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 request involves no significant hazards consideration.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: January 21, 1999.

Description of amendment request: The proposed amendments would change Technical Specification Tables 3.3.6.1–1 and 3.3.6.2–1 by increasing the Allowable Values for the high radiation trip for the exhaust monitors for the reactor building and the refueling. The January 21, 1999, amendment request supercedes the July 22, 1998, amendment request which was noticed in the **Federal Register** on August 26, 1998 (63 FR 45529).

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

The Unit 1 and Unit 2 reactor building and refueling floor ventilation exhaust radiation monitors perform no function in preventing, or decreasing the probability of, a previously evaluated accident. The monitors are designed to monitor ventilation exhaust for indications of a release of radioactive material resulting from a design basis accident and initiate appropriate protective actions. Because the proposed changes affect only the ventilation exhaust radiation monitors, the probability of an accident previously evaluated remains the same.

The function of the reactor building and the refueling floor ventilation exhaust radiation monitors, in combination with other accident mitigation systems, is to limit fission product release during and following postulated design basis accidents. The proposed new Allowable Values for the high radiation trip will continue to ensure the offsite doses resulting from a design basis accident do not exceed the NRC-approved licensing basis. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes increase the radiation level at which the ventilation exhaust monitors actuate; however, the manner in which their actuation logic functions and the systems that isolate or actuate as a result are unaffected by the proposed changes. Furthermore, the ventilation exhaust monitors will continue to perform their design function of limiting offsite doses to NRC-approved licensing limits at the higher Allowable Values. Therefore, the proposed changes cannot create the possibility of a new or different kind of accident from any previously evaluated.

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

The Bases for Unit 1 and Unit 2 Technical Specifications Tables 3.3.6.1–1 and 3.3.6.2–1 state that the Allowable Values for the reactor building and refueling floor ventilation exhaust radiation monitors ''are chosen to ensure radioactive releases do not exceed offsite dose limits.'' The proposed Allowable Values ensure the radiation monitors actuate at a radiation level sufficient to ensure offsite doses are within the NRC-approved licensing basis. The proposed Allowable Values comply with the margin of safety defined in the Technical Specifications Bases for the ventilation exhaust radiation monitors; therefore, the proposed changes do not reduce a margin of safety.

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

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

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

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: March 30, 1999.

Description of amendment request: The licensee has proposed to relocate Technical Specification 3/4.3.3.4, ''Meteorological Instrumentation,'' and its associated Bases to the Technical Requirements Manual (TRM). Because the TRM is incorporated within the South Texas Project updated final safety analysis report (UFSAR) for the units, changes to the requirements on the meteorological instrumentation that would be relocated to the TRM would be controlled in accordance with 10 CFR 50.59.

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

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

The affected system and components [i.e., meteorological monitoring instrumentation] are not assumed as initiators of analyzed events, and are not assumed to mitigate accident or transient events. The requirements and surveillances for [this affected system] and components will be relocated from the Technical Specifications to the Technical Requirements Manual, which is incorporated in the South Texas Project UFSAR and will be maintained pursuant to 10 CFR 50.59. In addition, the Meteorological Monitoring System components are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. The associated changes to the Technical Specification Index are administrative. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. This change will not impose different requirements, and adequate control of information will be maintained. Furthermore, this change will not alter assumptions stated in the safety analysis or licensing basis. The associated changes to the Technical Specification Index are administrative. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change will not reduce a margin of safety because the change has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structures, systems, and components remain the same as the existing Technical Specifications. Because any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59; there is no [significant] reduction in a margin of safety. The associated changes to the Technical Specification Index are administrative and have no potential effect on the margin of safety. Therefore, this change does not involve a significant reduction in the margin of safety.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of application for amendments: March 2, 1999 (TS 98–05).

Brief description of amendments: The proposed amendments would change the SQN Operating Licenses DPR–77 (Unit 1) and DPR–79 (Unit 2) by eliminating a requirement to have an Independent Safety Engineering Group (ISEG), conditions imposed by NUREG–0737. Because of evolution through numerous reorganizations and reassignments, these license conditions are no longer necessary and the Tennessee Valley Authority (TVA, the licensee) proposes deleting them.

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

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

The possibility of occurrence or the consequences for an accident or malfunction of equipment is not increased. The ISEG function is one of ''oversight'' only.

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

A possibility for an accident or malfunction of a different type than any evaluated previously in SQN's Final Safety Analysis Report is not created by the proposed elimination of the ISEG; nor is the possibility for an accident or malfunction of a different type. The ISEG function is one of ''oversight'' only.

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

The proposed amendment will not involve a significant reduction in the margin of safety. The ISEG function is one of ''oversight'' only.

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

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

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

NRC Section Chief: Sheri R. Peterson.

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

Date of amendment request: February 12, 1999 (TXX–99022).

Brief description of amendments: The proposed changes would modify the steam generator tube inspection requirements and acceptance criteria to implement the 1.0-volt repair criteria for steam generator tubes affected by outer diameter stress corrosion cracking (ODSCC) according to Nuclear Regulatory Commission (NRC) Generic Letter 95–05 (''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking'') at Comanche Peak Unit 1. Also proposed is the use of a voltage-dependent probability of detection; the methodology was originally submitted to the NRC by the Nuclear Energy Institute in 1996.

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

(1) Operation of Comanche Peak Unit 1 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Tube burst criteria are inherently satisfied during normal operating conditions due to the proximity of the tube support plate [TSP]. Test data indicates that tube burst cannot occur within the TSP, even for tubes which have 100% through-wall electric discharge machining notches, 0.75 inch long, provided that the TSP is adjacent to the notched area. Since tube to tube support plate proximity precludes tube burst during normal operating conditions, use of the criteria must retain tube integrity characteristics which maintain a margin of safety of 1.43 times the bounding faulted condition (Steam Line Break) pressure differential. As previously stated, the RG [Regulatory Guide] 1.121 criterion requiring maintenance of a safety factor of 1.43 times the Steam Line Break pressure differential on tube burst is satisfied by ¾″ diameter tubing with bobbin coil indications with signal amplitudes less than 4.7 volts, regardless of the indicated depth measurement. At the FDB [flow distribution baffle], a safety factor of 3 against the normal operating condition at power is applied. Here a voltage of 3.34 volts satisfies the burst capability recommendation.

The upper voltage repair limit (V_{URL}) will be determined prior to each outage using the most recently approved NRC database to determine the tube structural limit (V_{SL}). The structural limit is reduced by allowances for nondestructive examination (NDE) uncertainty (V_{NDE}) and growth (V_{Gr}) to establish V_{URL}. As an example, the NDE uncertainty component of 20% and a voltage growth allowance of 30% per full power year can be utilized to establish a V_{URL} of 3.13 volts for TSP indications, 2.22 volts for the FDB indications. The 20% NDE uncertainty represents a squareroot-sum-of-the-squares (SRSS) combination of probe wear uncertainty and analyst variability.

The flaw growth allowance should be an average growth rate or 30% per effective full power year, whichever is larger. The 30% growth allowance used to determine V_{URL} is conservative for the current conditions at Comanche Peak Unit 1. The average growth of the bobbin indication voltages observed at the last inspection is determined to be 0.14 volts, or 24.6% voltage growth. This value is a conservative representation of the growth trends at Comanche Peak Unit 1 as not all steam generators were inspected at end of cycle 3 and end of cycle 4, and the largest reported voltage growths represent more than one cycle of actual plant operation. The most current NRC approved database, contained in EPRI [Electric Power Research Institute] NP–7480–L, Addendum 1, was used to establish the V_{URL} values for the FDB and TSP intersections. Once approved by the NRC, the industry protocol for updating the database will be followed by TU Electric, ensuring that the most current database is utilized for all future applications of the criteria.

Also, assuming the criteria was applied at the last inspection at Comanche Peak Unit 1, using conservative growth projections as described in Reference 2 [of the February 12, 1999, application], the conditional burst probability at end of cycle 6 is determined to be 1.7×10^{-4}, which is well within the GL 95–05 reporting limit of 1×10^{-2}.

Relative to the expected leakage during accident condition loadings, it has been previously established that a postulated main Steam Line Break outside of containment but upstream of the MSIV [main steam isolation valve] represents the most limiting radiological condition relative to the plugging criteria. In support of implementation of the revised plugging limit, it will be determined whether the distribution of cracking indications at the tube support plate intersections during future cycles are projected to be such that primary to secondary leakage would result in site boundary doses within 10CFR100 guidelines and control room doses within the GDC [General Design Criterion]-19 limit. A separate calculation has determined this allowable Steam Line Break leakage limit to be 27.79 gpm in the faulted loop assuming a RCS [reactor coolant system] dose equivalent I–131 concentration of 1.0 microCi/gm. The establishment of the 27.79 gpm leak rate value is controlled by the 0 to 2 hour offsite dose at the site boundary for the accident initiated iodine spike case, not the control room dose. For this case, the site boundary thyroid dose approaches, but is bounded by, the 30 Rem limit recommended in NUREG–0800 [''Standard Review Plan''].

The methods for calculating the radiological dose consequences are also revised for this application. Rather than basing the calculated thyroid dose consequences on conversion factors from TID–14844, [''Calculation of Distance Factors for Power and Test Reactor Sites''] factors obtained from ICRP–30 [International Commission on Radiation Protection Publication 30] are used. The use of ICRP–30 dose conversion factors in this application has been previously accepted by the NRC. Although the use of ICRP–30, relative to the TID–14844, results in lower calculated thyroid doses for this application, the NRC has previously determined that the ICRP–30 factors retain adequate conservatism.

In summary, due to the methodology used to determine the maximum allowable, accident-initiated leak rate (prescribed in Section 2.b.4 of Generic Letter 95–05), the calculated radiological consequences at the EAB [exclusion area boundary] and LPZ [low population zone] are larger than previously reported for the postulated steamline break event. However, the calculated radiological consequences remain in compliance with NUREG–0800 and GDC–19. Therefore, it is concluded that the proposed changes do not result in a significant increase in the radiological consequences of an accident previously analyzed.

The removal from the FSAR [final safety analysis report] of the steamline break radiological dose consequences calculation typically identified as a ''5% failed fuel'' scenario does not affect the probability or consequences of any accident previously considered. For CPSES [Comanche Peak Steam Electric Station], no accident-induced fuel failures are predicted; therefore, consistent with NUREG–0800, this scenario is not required to be analyzed or presented in the FSAR.

In summary, because the implementation of the 1.0 volt voltage-based plugging criteria at Comanche Peak Unit 1 does not adversely affect steam generator tube integrity and implementation will be shown to result in acceptable radiological dose consequences, the proposed Technical Specification change does not result in any increase in the probability or consequences of an accident previously evaluated within the Comanche Peak FSAR.

(2) The proposed license amendment does not create the possibility of a new or different

kind of accident from any accident previously evaluated.

Implementation of the proposed steam generator tube 1.0 volt plugging limit does not introduce any significant changes to the plant design basis. Neither a single or multiple tube rupture event would be expected in a steam generator in which the plugging limit has been applied (during all plant conditions).

The bobbin probe voltage-based tube plugging criteria of 1.0 volt is supplemented by: enhanced eddy current inspection guidelines to provide consistency in voltage normalization, a 100% eddy current inspection sample size at the tube support plate elevations, and RPC [rotating pancake coil] inspection requirements for the larger indications left in service to characterize the principal degradation as ODSCC. TU Electric will implement a maximum normal operating condition primary to secondary leakage rate limit of 150 gpd (0.1 gpm—at room temperature) per steam generator to help preclude the potential for excessive leakage during all plant conditions. The 150 gpd leakage limit is more restrictive than the standard operating leakage limit (of 500 gpd) and is intended to provide additional margin to accommodate a stress corrosion crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Leakage trending capability consistent with EPRI Report TR–04788, ''PWR Primary-to-Secondary Leak Guidelines'', has been implemented at Comanche Peak Unit 1.

As steam generator tube integrity upon implementation of the 1.0 volt plugging limit continues to be maintained through in-service inspection and primary to secondary leakage monitoring, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

The use of the voltage-based bobbin probe tube support plate elevation plugging criteria at Comanche Peak Unit 1 maintains steam generator tube integrity commensurate with the criteria of Regulatory Guide 1.121. Regulatory Guide 1.121 describes a method acceptable to the NRC staff for meeting GDCs 14, 15, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the proposed criteria, even under the worst case conditions, the occurrence of ODSCC at the tube support plate elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The end of cycle distribution of crack indications at the tube support plate elevations is confirmed to result in acceptable primary to secondary leakage during all plant conditions and that radiological consequences are not adversely impacted.

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

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

Attorney for licensee: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm

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: February 16, 1999.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) Sections 3.6, 3.9, and 3.16 and the associated Bases for those sections for Units 1 and 2. The proposed changes would consolidate the auxiliary feedwater (AFW) cross-connect requirements by relocating the electrical power requirements from Section 3.16 to Section 3.6. The proposal also would clarify the TS with regard to permitting simultaneous entry into certain conditions of operation on Units 1 and 2.

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

Criterion 1—Operation of Surry Units 1 and 2 in accordance with the proposed TS change does not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated.

The proposed TS change is administrative in nature, and station operations are not being affected. The accidents considered relative to this proposed TS change are Rupture of Main Steam Pipe, Loss of All AC Power, and Loss of Feedwater. The probability of occurrence of these accidents has been previously evaluated to support Surry TS Amendment 143/140. The NRC reviewed the PSA [probabilistic safety analysis] basis during issuance of TS Amendment 143/140 and found it acceptable. The probability of occurrence of these accidents has been recently reviewed relative to this proposed TS change. It has been concluded that the proposed TS change is consistent with the existing analyses and evaluations and, therefore, will not increase the probability of occurrence of the identified accidents.

The consequences of the accidents identified above were also previously evaluated to support Surry TS Amendment 143/140. The PSA considerations included the AFW cross-connect capability, diesel generator dependencies, various LCO [limiting condition for operation] time periods, and a HELB [high energy line break] in the vicinity of the AFW Pumps. The previous evaluation was recently reviewed relative to this proposed TS change. This review determined that the proposed TS change is consistent with the design and licensing bases supporting the existing Technical Specifications. The proposed TS change is also consistent with the existing analyses and evaluations, the consequences of which bound any potential consequences of the proposed TS change. Therefore, the proposed TS change will not increase the consequences of the identified accidents.

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

The possibility for a new or different type of accident than any previously evaluated is not created since the considerations in the PSA and evaluations performed to support TS Amendment 143/140 are not changed by the proposed administrative TS change. The proposed TS change is consistent with the design and licensing bases supporting the existing Technical Specifications. Furthermore, station operations and plant equipment are not being affected and, therefore, the proposed TS change does not create any new failure modes or accident precursors.

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

The proposed administrative change to Surry Technical Specifications clarifies the requirements (limiting conditions for operation (LCO) and action statements) relating to the Auxiliary Feedwater (AFW) cross-connect by relocating the emergency power source requirements of TSs 3.16.A.8 and 3.16.B.4 to TS 3.6. The proposed TS change does not alter the current TS requirements or bases, as well as maintains the Surry licensing and design basis. The proposed change does not affect either station operations or plant equipment, hence the availability of equipment for the mitigation of accidents is not decreased. Furthermore, the assumptions governing the accident analyses remain unchanged, and the consequences of the existing analyses and evaluations remain bounding. This is an administrative change and as such does not involve a significant reduction in a margin of safety.

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

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

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.

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: February 16, 1999.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) Section 4.2 for Units 1 and 2 to relax the surveillance requirements for reactor coolant pump (RCP) flywheels. The flywheels provide extended reactor coolant flow coastdown capability if electric power for the RCPs is lost. Currently, the flywheels are subjected to an inspection program that meets the requirements of NRC Regulatory Guide 1.14, Revision 1, dated August 1975. The inspections include an ultrasonic examination (UT) of areas of high stress concentration at the bore and keyway every three years, and complete UT every 10 years. The proposed change would require only a 10-year UT, based upon an analysis presented in a Westinghouse topical report (WCAP–14535A) which has been reviewed and accepted by NRC staff.

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

a. The reduction of the inspection requirements for the reactor coolant pump flywheels, as generically approved by the NRC and technically supported by WCAP–14535A, does not significantly increase the probability of an accident previously evaluated in the safety analysis report. The results of WCAP–14535A have been reviewed and evaluated with the technical basis accepted for referencing in license applications by the NRC in their letter entitled ''Acceptance for referencing of Topical Report WCAP–14535, Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination,'' dated September 12, 1996.

The proposed Technical Specification change reduces the surveillance requirements (inspection) on the RCP flywheel. There is no change in the method of plant operation or system design. The WCAP–14535A report establishes that the proposed change has a negligible affect on the probability that the flywheel will fail given that the flywheels received preservice and inservice examinations as required previously. Therefore, the proposed change does not increase the probability of occurrence or consequences of any previously analyzed accident.

b. The proposed change to reduce the inspection requirements for the RCP flywheels as generically approved by the NRC and supported by WCAP–14535A does not create the possibility of a new or different kind of accident from any accident previously evaluated in the safety analysis report.

The proposed surveillance requirements (inspection) only reduce the inspection requirements/frequency for the reactor coolant pump flywheels, and there is no change in the method of plant operation or system design.

c. The proposed change reducing the inspection of the RCP flywheels as generically approved by the NRC and supported by WCAP–14535A, does not impact the accident analysis assumptions or the basis of any Technical Specification. As previously stated, the analysis performed in the WCAP–14535A report established that the affect on flywheel failure probability was negligible given that the initial preservice and inservice inspections under the current requirements were performed. Therefore, the proposed change in surveillance (inspection) frequency does not involve a significant reduction in the margin of safety.

The analysis provided herein demonstrates that the proposed amendment to the Surry Technical Specifications does not involve a significant increase in the probability or consequences of a previously evaluated accident, does not create the possibility of a new or different kind of accident, and does not involve a significant reduction in a margin of safety.

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

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

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.

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

Date of amendment request: April 12, 1999 (TSCR 212).

Description of amendment request: The purpose of the proposed amendments is to update references in the Technical Specifications. The update is necessary to reflect relocation of referenced information in the Final Safety Analysis Report.

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

1. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not create a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment corrects references within the Technical Specification requirements such that they refer to the correct information in the updated Final Safety Analysis Report (FSAR). The references changed due to relocation of the information within the FSAR. The Technical Specification requirements and intent are not changed. Therefore, these changes are administrative only and do not change the design or operation of the Point Beach Nuclear Plant [PBNP]. Operation of PBNP in accordance with the proposed amendments cannot increase the probability or consequences of an accident previously evaluated.

2. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not create the possibility of a new or different kind of accident previously evaluated.

The proposed changes are administrative only and therefore do not materially change any requirements for the design or operation of PBNP. Therefore, operation in accordance with the proposed changes cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not create a significant reduction in a margin of safety.

The proposed changes are administrative only; correcting references within the Technical Specification requirements. No requirement on the operation or design of the facility is being changed. Therefore, there is no reduction in the margin of safety.

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

Local Public Document Room location: The Lester Public Library, 1001 Adams Street, Two Rivers, Wisconsin 54241.

Attorney for licensee: John H. O'Neill, Jr., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: George F. Dick, Jr., Acting.

Notice of Issuance of Amendments to Facility Operating Licenses

During the period since publication of the last biweekly notice, the

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

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

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

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

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

Date of application for amendments: November 19, 1998.

Brief description of amendments: The amendments revised Technical Specification 3.7.6 ''Service Water (SRW) System'' to allow operation of Calvert Cliffs Unit Nos. 1 and 2 with one SRW plate and frame heat exchanger in a subsystem secured and removing one containment air cooler from service to enable the affected SRW subsystem to remain operable.

Date of issuance: April 14, 1999.

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

Amendment Nos.: 230 and 206.

Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: December 16, 1998 (63 FR 69333). The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated April 14, 1999.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: November 1, 1996, as supplemented May 22, 1998, September 14, 1998, January 4, 1999, and March 19, 1999.

Brief description of amendment: The amendment modified the Technical Specifications for the Brunswick Steam Electric Plant, Units 1 and 2, to extend the Allowed Outage Time for 4.16kV AC balance of plant buses and the AC electrical power distribution system load group buses.

Date of issuance: April 15, 1999.

Effective date: April 15, 1999.

Amendment Nos.: 205 and 235.

Facility Operating License Nos. DPR–71 and DPR–62: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: February 11, 1998 (63 FR 6977). The supplemental submittals of May 22, 1998, September 14, 1998, January 4, 1999, and March 19, 1999, contained clarifying information only, and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 15, 1999.

No significant hazards consideration comments received: No.

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

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: October 14, 1998.

Brief description of amendment: The amendment modifies the acceptance criterion for Surveillance Requirement 3.4.14.2 from the setpoint value of 465 psig to the analytical limit for the residual heat removal system of 474 psig reactor coolant system pressure.

Date of issuance: April 20, 1999.

Effective date: April 20, 1999.

Amendment No. 182.

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

Date of initial notice in Federal Register: November 4, 1998 (63 FR 59587).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 20, 1999.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: September 3, 1997, as supplemented March 13, 1998, and March 18, 1999.

Brief description of amendment: The amendment revises the technical specifications to delete snubber operability requirements, action requirements for inoperable snubbers, and snubber testing requirements. The snubber testing requirements have been relocated to the Palisades Operating Requirements Manual.

Date of issuance: April 13, 1999.

Effective date: April 13, 1999, and shall be implemented within 60 days.

Amendment No.: 185.

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

Date of initial notice in Federal Register: April 8, 1998 (63 FR 17222). The March 18, 1999, submittal requested a 60-day allowance for implementation of the amendment. This change was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 13, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Van Wylen Library, Hope College, Holland, Michigan 49423–3698.

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

Date of amendment request: July 17, 1998 (Reference NRC–98–0044).

Brief description of amendment: This amendment revises the Enrico Fermi

Atomic Power Plant, Unit 1, License to allow possession of a nominal amount of special nuclear material.

Date of issuance: April 15, 1999.

Effective date: On the date of issuance of this amendment and must be fully implemented no later than 60-calendar days from the date of issuance.

Amendment No.: 16.

Facility Operating License No. DPR–9: Amendment revised the License by adding new Part 2.B.4 to the License.

Date of initial notice in Federal Register: October 21, 1998 (63 FR 56240). The NRC's related evaluation of the amendment is contained in a Safety Evaluation dated April 15, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161.

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

Date of application for amendments: March 15, 1999, and supplemented by letter dated March 17, 1999.

Brief description of amendments: The amendments delete from the joint Technical Specifications Section 3.3.7, ''Control Room Area Ventilation System (CRAVS) Actuation Instrumentation,'' and Section 3.3.8, ''Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) Actuation Instrumentation.'' These surveillance requirements are not applicable to Catawba because the sections do not reflect the design of the Catawba units.

Date of issuance: April 8, 1999.

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

Amendment Nos.: Unit 1—177; Unit 2—169.

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

Date of initial notice in FEDERAL REGISTER: March 24, 1999 (64 FR 14274). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 8, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 18, 1999.

Brief description of amendments: The amendments revise the Technical Specifications Surveillance Requirement (SR) 3.6.16.1 regarding surveillance of reactor building access openings, SR 3.6.16.3 regarding surveillance of reactor building structural integrity, and Administrative Controls 5.5.2 regarding the Containment Leakage Rate Testing Program. The revised requirements would provide scheduling flexibility without decreasing quality and safety margin.

Date of issuance: April 9, 1999.

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

Amendment Nos.: 178—Unit 1; 170—Unit 2.

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

Date of initial notice in Federal Register: March 10, 1999 (64 FR 11961). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 9, 1999.

No significant hazards consideration comments received: No.

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

Duquesne Light Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of application for amendment: November 11, 1998, as supplemented February 26, 1999.

Brief description of amendment: The amendment modified License Condition 2.C(9) to allow, on a one-time only, extension of the steam generator inspection interval in Technical Specification Surveillance 4.4.5.3.b. This will allow the steam generator inspection interval to coincide with the thirteenth refueling outage or the end of 500 effective full power days, whichever occurs sooner.

Date of issuance: April 16, 1999.

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

Amendment No: 221.

Facility Operating License No. DPR–66. Amendment revised the License.

Date of initial notice in Federal Register: December 2, 1998 (63 FR 66593). The February 26, 1999, letter provided additional information but did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the initial notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 16, 1999.

No significant hazards consideration comments received: No.

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

Entergy Operations, Inc., Docket Nos. 50–313 and 50–368, Arkansas Nuclear One, Units 1 and 2, Pope County, Arkansas

Date of amendment request: June 28, 1996, as supplemented by letters dated February 23 and March 15, 1999.

Brief description of amendments: The amendments revise the Technical Specifications to permit the containment equipment hatch to be open during handling of irradiated fuel in containment and core alterations provided that the capability for closure is maintained.

Date of issuance: April 16, 1999.

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

Amendment Nos.: Unit 1—Amendment No. 195; Unit 2—Amendment No. 203.

Facility Operating License Nos. DPR–51 and NPF–6: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 14, 1996 (61 FR 42280). The February 23 and March 15, 1999, letters provided clarifying information that did not change the scope of the original application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 16, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 30, 1998.

Brief description of amendment: The amendment revises the single largest post-accident load capable of being supplied by the diesel generators and relocates this value to the Bases for Technical Specification (TS)

Surveillance 4.8.1.1.2.c.3. TS Surveillance 4.8.1.1.2.c.3 has been revised to refer to ''the single largest post-accident load'' rather than a specific numerical value for diesel generator load reject testing. This change is consistent with the guidance provided in NUREG–1432 , ''Improved Standard Technical Specifications for Combustion Engineering Plants.''

Date of issuance: April 21, 1999.

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

Amendment No.: 204.

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

Date of initial notice in Federal Register: October 21, 1998 (63 FR 56241). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 21, 1999.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: November 13, 1997.

Brief description of amendment: The amendment changes the Appendix A Technical Specifications (TSs) by revising TS 6.8.4.a, Primary Coolant Sources Outside Containment, to add portions of the containment vacuum relief and primary sampling systems to the list of systems included in the Primary Coolant Sources Outside Containment Program.

Date of issuance: April 21, 1999.

Effective date: The license amendment is effective as of its date of issuance, and shall be implemented within 60 days.

Amendment No.: 150.

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

Date of initial notice in Federal Register: February 25, 1998 (63 FR 9601). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 21, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 1, 1997, as supplemented April 23 and November 17, 1998 and February 19, 1999.

Brief description of amendment: The changes specify criteria for evaluating the growth of pit-like intergranular attack steam generator tube degradation identified in tubes in the ''B'' once-through steam generator (OTSG). Florida Power Corporation also requested to amend the Improved Technical Specifications to clarify the date by which the OTSG inservice inspection results are required to be submitted to the NRC.

Date of issuance: April 8, 1999.

Effective date: April 8, 1999.

Amendment No.: 172.

Facility Operating License No. DPR–72: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: October 22, 1997 (62 FR 54873). The supplemental letters dated April 23 and November 17, 1998, and February 19, 1999 did not change the original no significant hazards determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 8, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428.

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

Date of application for amendment: September 9, 1997, as supplemented November 7 and 25, 1997, and January 20 and October 30, 1998.

Brief description of amendment: The amendment proposed to revise the Final Safety Analysis Report (FSAR) analysis of the Makeup System letdown line failure accident. The revised analysis models the event as being terminated by manual operator action to isolate the line whereas the original analysis models an automatic isolation of the break.

Date of issuance: April 13, 1999.

Effective date: April 13, 1999.

Amendment No.: 173.

Facility Operating License No. DPR–72: Amendment approves changes to the Final Safety Analysis Report.

Date of initial notice in Federal Register: September 24, 1997 (62 FR 50005). The supplemental letters dated November 7 and 25, 1997, January 20, 1998, and October 30, 1998, did not change the original proposed no significant hazards consideration determination, or expand the scope of the amendment request as originally noticed.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 13, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428.

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

Date of application for amendment: October 30, 1998, as supplemented April 7, 1999.

Brief description of amendment: Changes the Crystal River Unit 3 Technical Specifications to delete a note regarding the number of required channels for the Degrees of Subcooling function, and to subdivide the Core Exit Temperature (Backup) function into two new functions in Table 3.3.17–1, Post-Accident Monitoring Instrumentation.

Date of issuance: April 20, 1999.

Effective date: April 20, 1999.

Amendment No.: 174.

Facility Operating License No. DPR–72: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: January 13, 1999 (64 FR 2246). The April 7, 1999, supplement did not affect the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 20, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428.

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

Date of application for amendment: May 27, 1998, as supplemented October 9, 1998.

Brief description of amendment: Deletes the requirement for operability of the safety injection tanks in Mode 4 of reactor operation.

Date of Issuance: April 8, 1999.

Effective Date: Amendment is effective within 30 days of receipt.

Amendment No.: 100.

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

Date of initial notice in Federal Register: July 29, 1998 (63 FR 40556). The October 9, 1998 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 8, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: November 25, 1998, as supplemented February 12, 1999.

Brief description of amendment: The amendment approves the proposed surveillance Technical Specifications related to the once through steam generator inservice inspections to be completed during the 13R refueling outage in fall 1999. Related TS Bases changes are also included.

Date of issuance: April 13, 1999.

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

Amendment No.: 209.

Facility Operating License No. DPR–50. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: December 16, 1998 (63 FR 69342).

The February 12, 1999, submittal modified the request, but did not affect the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 13, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 15, 1998, as supplemented February 3, and February 12, 1999.

Brief description of amendment: The amendment authorizes a revision to the TMI–1 updated final safety analysis report (UFSAR) for use of revised atmospheric dispersion factors (X/Q) (obtained by utilizing recent meteorological data) in determining Chapter 14 postulated accident analysis radiological dose consequences at Technical Specification Section 5.1.1 defined exclusion area boundary (EAB) and low population zone (LPZ).

Date of issuance: April 15, 1999.

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

Amendment No.: 210.

Facility Operating License No. DPR–50. Amendment authorizes changes to the UFSAR.

Date of initial notice in Federal Register: November 18, 1999 (63 FR 64117).

The February 3, and February 12, 1999, letters were within the scope of the original application and did not change the staff's no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 15, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: December 28, 1998, as supplemented March 1 and 29, 1999.

Brief description of amendment: The amendment revises Technical Specification (TS) 2.2.1, ''Limiting Safety System Settings-Reactor Trip Setpoints,'' to reflect revised loss of normal feedwater flow analyses.

Date of issuance: April 8, 1999.

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

Amendment No.: 232.

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

Date of initial notice in Federal Register: February 10, 1999 (64 FR 6701). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 8, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: January 18, 1999.

Brief description of amendment: The amendment revises Technical Specification (TS) 3.6.1.2, ''Containment Systems—Containment Leakage,'' and also revises the related TS bases and Final Safety Analysis Report sections. The revisions relate to changes in the secondary containment bypass leakage.

Date of issuance: April 14, 1999.

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

Amendment No.: 234.

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

Date of initial notice in Federal Register: February 10, 1999 (64 FR 6703). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 14, 1999.

No significant hazards consideration comments received: No.

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

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 10, 1999.

Brief description of amendment: The amendment incorporates alternative inspection requirements into Technical Specification Surveillance Requirement 3/4.4.10, ''Structural Integrity,'' for the reactor coolant pump flywheel.

Date of issuance: April 16, 1999.

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

Amendment No.: 169.

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

Date of initial notice in Federal Register: March 10, 1999 (64 FR 11964).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 16, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 5, 1999, as supplemented March 1, 1999.

Brief description of amendments: The amendments revise certain requirements for repair of defective steam generator tubes specified in Technical Specification 4.12, ''Steam Generator Tube Surveillance,'' based on the latest revision to a previously approved methodology.

Date of issuance: April 15, 1999.

Effective date: April 15, 1999, with full implementation within 30 days.

Amendment Nos.: 144 Unit 1—135 Unit 2.

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

Date of initial notice in Federal Register: March 10, 1999 (64 FR 11964). The March 1, 1999, supplement provided corrected Technical Specification pages. This information was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 15, 1999.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: March 18, 1998.

Brief description of amendment: The amendment revises Technical Specification (TS) 5.2.f and TS 5.11.2 to change the title of ''Shift Supervisor'' to ''Shift Manager.''

Date of issuance: April 15, 1999.

Effective date: April 15, 1999.

Amendment No.: 190.

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

Date of initial notice in **Federal Register**: April 8, 1998 (63 FR 17227).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 15, 1999.

No significant hazards consideration comments received: No.

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

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: May 16, 1996.

Brief description of amendment: The amendment revises requirements for Plant Operating Review Committee review of fire protection program and procedure changes.

Date of issuance: April 12, 1999.

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

Amendment No.: 252.

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

Date of initial notice in Federal Register: July 3, 1996 (61 FR 34895).

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: December 16, 1998, as supplemented March 22, 1999.

Brief description of amendment: This amendment revised Technical Specification (TS) Surveillance Requirements 4.8.1.1.2 and 4.8.1.1.3, Table 4.8.1.1.2–1, and the associated Bases. These changes removed the emergency diesel generator accelerated testing and special reporting requirements from the TSs in accordance with the guidance provided in Generic Letter 94–01.

Date of issuance: April 14, 1999.

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

Amendment No.: 119.

Facility Operating License No. NPF–57: This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: January 13, 1999 (64 FR 2251).

The supplemental letters 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, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070.

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

Date of application for amendment: June 12, 1998, as supplemented July 23, 1998 and September 8, 1998.

Brief description of amendment: The amendment revises Technical Specification (TS) Limiting Condition for Operation Sections 3.7.1.1, 3.7.1.2, and 3.7.1.3. Specifically, the changes revise the Ultimate Heat Sink limits for river water temperature, in order to increase operational flexibility. In addition, the Station Service Water System (SSWS) and Safety Auxiliaries Cooling System (SACS) TS Action Statements have been revised to provide additional restrictions on continued plant operation. These revisions provide more explicit TS direction for plant operation under limiting SSWS/SACS configurations.

Date of issuance: April 19, 1999.

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

Amendment No.: 120.

Facility Operating License No. NPF–57: This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 1, 1999 (63 FR 35995) The July 23, 1998, and September 8, 1998, supplements 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 19, 1999.

No significant hazards consideration comments received: No

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070.

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

Date of application for amendment: September 18, 1998, as supplemented by letter dated February 5, 1999.

Brief description of amendment: The amendment revises Virgil C. Summer Nuclear Station Technical Specifications to permit use of the BEACON system. BEACON is a core power distribution monitoring and support system based on a three-dimensional nodal code.

Date of issuance: April 9, 1999.

Effective date: April 9, 1999.

Amendment No.: 142.

Facility Operating License No. NPF–12: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: November 18, 1998 (63 FR 64121).

The February 5, 1999, submittal contained clarifying information only, and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 9, 1999.

No significant hazards consideration comments received: No

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

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

Date of application for amendments: January 24, 1997.

Brief description of amendments: The amendments revised Surveillance Requirement (SR) 3.8.1.9 to Technical Specification 3.8.1, ''AC Sources—Operating,'' to more accurately reflect test conditions and plant design requirements.

Date of issuance: April 9, 1999.

Effective date: April 9, 1999, to be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 2–151; Unit 3–143.

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

Date of initial notice in Federal Register: February 11, 1998 (63 FR 6997).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 9, 1999.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: January 20, 1999.

Brief description of amendments: The amendments revise the descriptive details of Technical Specification 4.7.1.2.1.a, regarding performance testing of the Auxiliary Feedwater (AFW) pumps, to more closely adhere to NUREG–1431, ''Improved Standard Technical Specifications for Westinghouse Plants.'' This involves relocating the surveillance-required numerical values for the AFW pump performance test discharge pressure and flow rate to the South Texas Project Updated Final Safety Analysis Report.

Date of issuance: April 16, 1999.

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

Amendment Nos.: Unit 1—Amendment No. 105; Unit 2—Amendment No. 92.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: February 24, 1999 (64 FR 9201).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 16, 1999.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: January 26, 1999.

Brief description of amendments: The amendments revise part of the inservice inspection requirements for the reactor coolant pump flywheel from an in-place ultrasonic volumetric examination of the areas of higher stress concentration at the bore and keyway at approximately 3-year intervals and a surface examination of all exposed surfaces and complete ultrasonic volumetric examination at approximately 10-year intervals to ultrasonic examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius once every 10 years.

Date of issuance: April 16, 1999.

Effective date: April 16, 1999, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 1—Amendment No. 106; Unit 2—Amendment No. 93.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: March 10, 1999 (64 FR 11968).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 16, 1999.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: September 30, 1998.

Brief description of amendments: Revises Units 1 and 2 Technical Specification (TS) Section 3/4.4.5, ''Steam Generator'' Surveillance Requirements. The future installation of the new Delta 94 steam generators at the South Texas Project, Units 1 and 2 necessitates changes to the steam generator tube sample selection and inspection requirements; inservice inspection frequencies; acceptance criteria; and inspection reporting requirements.

Date of issuance: April 19, 1999.

Effective date: April 19, 1999, to be implemented following the replacement of Unit 1 Model E steam generators with Model delta94 steam generators and prior to Unit 1 operation with the delta94 steam generators installed.

Amendment Nos.: Unit 1—Amendment No. 107; Unit 2—Amendment No. 94.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: November 4, 1998 (63 FR 59595).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 19, 1999.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: August 6, 1997, as supplemented by letters dated September 4 and 18, 1997, December 9, 1997, and February 4, 1999.

Brief description of amendments: The amendments revise Technical Specification (TS) Table 2.2–1 and TS 3/4.2.5 to allow the reactor coolant system total flow rate to be determined using cold leg elbow tap differential pressure measurements.

Date of issuance: April 19, 1999.

Effective date: As of its date of issuance to be implemented within 7 days of issuance.

Amendment Nos.: Unit 1—Amendment No. 108; Unit 2—Amendment No. 95.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 14, 1997 (62 FR 43556).

The September 4 and 18, 1997, December 9, 1997, and February 4, 1999, letters provided clarifying information that did not change the original application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 19, 1999.

No significant hazards consideration comments received: No.

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

Tennessee Valley Authority, Docket No. 50–328, Sequoyah Nuclear Plant, Unit 1, Hamilton County, Tennessee

Date of application for amendments: August 27, 1998, supplemented by letter dated March 19, 1999 (TS 98–04).

Brief description of amendments: The amendments change the Technical Specifications (TS) for Sequoyah Nuclear Plant, Unit 2 reactor by adding a sentence at the end of TS Section 5.3 authorizing installation of a limited number of lead test assemblies containing downblended uranium in accordance with Topical Report BAW–2328.

Date of issuance: April 12, 1999.

Effective date: April 12, 1999.

Amendment Nos.: 234.

Facility Operating License No. DPR–79: The amendment revises the TS.

Date of initial notice in Federal Register: March 10, 1999 (64 FR 11969). The supplemental letter of March 19, 1999 did not change the initial proposed no significant hazards condition determination.

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

No significant hazards consideration comments received: None.

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

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

Date of application for amendment: June 29, 1998, as supplemented by letter dated February 19, 1999.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.7.1.7 operability requirements to require four atmospheric steam dump (ASD) lines to be operable. Other changes were made to TS 3.7.1.7 to address action statements and surveillance requirements for the four ASD lines.

Date of issuance: April 20, 1999.

Effective date: April 20, 1999, to be implemented within 30 days from the date of issuance.

Amendment No.: 131.

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

Date of initial notice in Federal Register: September 9, 1998 (63 FR 48271).

The February 19, 1999, supplemental letter provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 20, 1999.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: May 28, 1998, as supplemented December 11, 1998.

Brief description of amendments: These amendments revise Technical Specifications (TS) to provide a specific numerical setting for reactor trip, reactor coolant pump trip, and auxiliary feedwater initiation on a loss of power to the 4 kilovolt (kV) buses. Changes to the bases for the affected TS sections are also being made.

Date of issuance: April 23, 1999.

Effective date: April 23, 1999.

Amendment Nos.: Unit 1–189; Unit 2–194.

Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: July 15, 1998 (63 FR 38208).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 23, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: The Lester Public Library, 1001 Adams Street, Two Rivers, Wisconsin 54241.

Dated at Rockville, Maryland, this 28th day of April 1999.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 99–11119 Filed 5–4–99; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

DCI Telecommunication, Inc., File No. 500–1; Order of Suspension of Trading

May 3, 1999.

It appears to the Securities and Exchange Commission (''Commission'') that there is a lack of current and accurate information concerning the securities of DCI Telecommunications, Inc. (''DCI'') because of questions regarding the accuracy and adequacy of DCI's financial statements, specifically, DCI's apparent inflation of revenues by accounting for one of more business combinations as a pooling of interests.

The Commission is of the opinion that the public interest and the protection of investors require a suspension of trading in the securities of DCI.

Therefore, it is ordered, pursuant to Section 12(k) of the Securities Exchange Act of 1934, that trading in DCI securities is suspended for the period from 9:30 a.m. EST, May 3, 1999 through 11:59 p.m. EST, on May 14, 1999.