appearance test in CTS 4.9.A.2.e.1.d) (Change ITS 5.5–M.4)

Environmental Impacts of the Proposed Action

The Commission has completed its evaluation of the proposed conversion of the CTS to the ITS for the CNS, including the six beyond-scope issues identified above. Changes which are administrative in nature have been found to have no effect on the technical content of the TS.

The increased clarity and understanding these changes bring to the TS are expected to improve the operators control of the CNS in normal and accident conditions.

Relocation of requirements from the CTS to other licensee-controlled documents does not change the requirements themselves. Future changes to these requirements may then be made by the licensee under 10 CFR 50.59 and other NRC-approved control mechanisms which will ensure continued maintenance of adequate requirements. All such relocations have been found consistent with the guidelines of NUREG–1433 and the Commission's Final Policy Statement.

Changes involving more restrictive requirements have been found to enhance station safety.

Changes involving less restrictive requirements have been reviewed individually. When requirements have been shown to provide little or no safety benefit, or to place an unnecessary burden on the licensee, their removal from the TS is justified. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of a generic action, or of agreements reached during discussions with the OG and found to be acceptable for the station. Generic relaxations contained in NUREG–1433 have been reviewed by the NRC staff and found to be acceptable.

In summary, the proposed revisions to the TS have been found to provide control of station operations such that reasonable assurance will be provided that the health and safety of the public will be adequately protected.

The proposed amendment will not increase the probability or consequences of accidents, will not change the quantity or types of any effluent that may be released offsite, and will not significantly increase occupational or public doses. Also, these changes do not affect the design of the station, do not involve any modifications to the station, and do not increase the licensed power and allowable effluents for the station. The changes will not create any new or unreviewed environmental impacts that were not considered in the Final Environmental Statement (FES) related to the operation of the CNS dated February 1973. Therefore, there are no significant radiological impacts associated with the proposed amendment.

With regard to potential non-radiological impacts, the proposed amendment involves features located entirely within the restricted area defined in 10 CFR Part 20. They do not affect non-radiological station effluents and have no other environmental impact. Therefore, there are no significant non-radiological environmental impacts associated with the proposed amendment.

Accordingly, the Commission concludes that there are no significant environmental impacts associated with the proposed amendment.

Alternatives to the Proposed Action

Since the Commission has concluded there is no significant environmental impact associated with the proposed amendment, any alternatives with equal or greater environmental impact need not be evaluated. The principal alternative to the proposed amendment would be to deny the amendment. Denial of the licensee's application would not reduce the environmental impacts of the CNS operations, but it would prevent the safety benefits to the station from the conversion to the ITS. The environmental impacts of the proposed action and the alternative action are similar.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the FES for the CNS.

Agencies and Persons Consulted

In accordance with its stated policy, on July 22, 1998, the staff consulted with the Nebraska State official, Cheryl Rogers of the State Department of Health, regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

Based upon the environmental assessment, the Commission concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the Commission has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's application dated March 27, 1997, as supplemented by the letters dated September 29 and December 22, 1997, and February 9, March 13, March 26, April 16, and May 6, 1998, 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 located at the Auburn Memorial Library, 1810 Courthouse Avenue, Auburn, Nebraska 68305.

Dated at Rockville, Maryland, this 23rd day of July 1998.

For the Nuclear Regulatory Commission.

David L. Wigginton,

Acting Director, Project Directorate IV–1, Division of Reactor Projects III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 98–20235 Filed 7–28–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from July 3, 1998, through July 17, 1998. The last biweekly notice was published on July 15, 1998 (63 FR 38198).

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 August 14, 1998, 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.

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

Date of amendment request: May 27, 1997, as supplemented by letters dated March 9, March 20, April 20, May 27, and June 24, 1998

Description of amendment request: The proposed amendments would revise the current Technical Specifications (TS) of each unit to conform with NUREG–1431, Revision 1, ''Standard Technical Specifications—Westinghouse Plants.'' The staff had previously issued a Notice of Consideration of Issuance of Amendments published in the **Federal Register** on July 14, 1997 (62 FR 37628) covering all the proposed changes that were indeed within the scope of NUREG–1431. The staff subsequently published two Notices of Consideration of Issuance of Amendments and Proposed No Significant Hazards Determination (63 FR 25106, dated May 6, 1998; 63 FR 27760 dated May 20, 1998) to cover DEC's March 9, March 20, April 20, and May 27, 1998, supplements, which proposed changes that are beyond the scope of NUREG–1431. On June 24, 1998, DEC identified additional beyond-scope changes. The following descriptions and proposed no significant hazard analyses cover only those beyond-scope changes. Associated with each change are administrative/editorial changes such that the new or revised requirements would fit into the format of NUREG–1431.

1. Current TS 4.8.1.1.2.f specifies that the fuel for the emergency diesel generators (EDGs) be periodically sampled for particulate contamination strictly in accordance with the industry standard ASTM-D2276–78. DEC proposed to relax this requirement, adopting only the guidance of the standard, but using a larger particulate filter for sampling (change from 0.8-to 3-micron). The revised requirement would show up as TS 5.5.13.c of the Improved TS. No changes to the design and functions of the EDGs are proposed.

2. DEC proposed to revise current TS Table 4.3–1, Functions 16 and 17. The revised requirements, to show up as Table 3.3.1–1, Functions 15 and 16.b, of the Improved TS, would add an actuation logic test surveillance for the reactor trip system interlocks and the safety injection input from the engineered safety feature actuation system. No changes to the design and functions of these systems are involved.

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

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

For all the changes the answer is ''no.'' The proposed changes will not affect the safety function of the subject systems. There will be no direct effect on the design or operation of any plant structures, systems, or components. No previously analyzed accidents were initiated by the functions of these systems, and the systems will continue to perform their functions in mitigating consequences of previously analyzed accidents. Therefore, the proposed changes will have no impact on the consequences or probabilities of any previously evaluated accidents.

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

For all the changes the answer is ''no.'' The proposed changes would not lead to any design or operating procedure change. Hence, no new equipment failure modes or accidents from those previously evaluated will be created.

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

For all the changes the answer is ''no.'' Margin of safety is associated with confidence in the design and operation of the plant. The proposed changes to the TS do not involve any change to plant design, operation, or analysis. Thus, the margin of safety previously analyzed and evaluated is maintained.

Based on this analysis, it appears that the three standards of 10 CFR 50.92(c) are satisfied for each of the proposed changes. 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: Mr. Paul R. Newton, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: September 15, 1997, as supplemented by letters dated March 5, April 27, and June 15, 1998.

Description of amendment request: The staff had previously published a Notice of Consideration of Amendments and Proposed No Significant Hazards Consideration Determination on the licensee's September 15, 1997, application in the **Federal Register** on October 8, 1997 (62 FR 52580). As a result of the staff's requests for additional information, DEC expanded its original amendment application by letter dated June 15, 1998. Specifically, the June 15, 1998, letter proposes requirements regarding the Low Temperature Overpressure Protection System to be added to the units' Technical Specifications. There is, however, no change to plant design.

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, addressing the three standards of 10 CFR 50.92(c):

First Standard

Implementation of this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. The Low Temperature Overpressure Protection System is not an accident initiating system; it is an accident mitigating system. Therefore, the addition of supplemental Technical Specification required controls pertaining to this system cannot impact accident initiating probabilities. The Low Temperature Overpressure Protection System will remain fully capable of performing its design accident mitigation function for the modes in which it is required. Therefore, no accident consequences will be impacted.

Second Standard

Implementation of this amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. As noted previously, the Low Temperature Overpressure Protection System is not an accident initiating system. The addition of the supplemental Technical Specification controls pertaining to this system as specified will not impact any plant systems that are accident initiators. No other modifications are being proposed to the plant which would result in the creation of new accident mechanisms.

Third Standard

Implementation of this amendment would not involve a significant reduction in a margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of the fission product barriers will not be impacted by implementation of this proposed amendment supplement. The Low Temperature Overpressure Protection System will remain fully capable of performing its design function for the modes in which it is required. Therefore, no safety margin will be significantly impacted.

The staff reviewed the licensee's analysis, and agrees that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the 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: Mr. Paul R. Newton, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Project Director: Herbert N. Berkow.

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: May 8, 1998.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) for the Power Range Neutron Flux High Trip setpoints in the event of inoperable main steam safety valves. The licensee has determined that the new values are more conservative than the values in the current TS. Also, the proposed changes would delete the references to the 3-loop operation. The proposed changes are consistent with the proposed Improved Standard TS submitted by the licensee on May 27, 1997.

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

1. Will operation of the facility in accordance with the proposed amendment involve an increase in the probability or consequences of an accident previously evaluated?

The proposed amendment involves a reduction in the maximum allowable power range neutron flux high setpoints in case of inoperable main steam safety valves. All applicable UFSAR [Updated Final Safety Analysis Report] Chapter 15 transient acceptance criteria are met with the proposed change. Therefore, operation of the facility in accordance with the proposed amendment will not involve an increase in the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated:

No new equipment or operating practice is involved with this proposed amendment. No alteration to any existing hardware is involved with this proposed amendment. Power Range high neutron flux setpoint calibration is continued to be performed by the same approved procedure. Therefore, operation of the facility in accordance with the proposed amendment will not create the possibility of any new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with the proposed amendment involve a reduction in a margin of safety?

The proposed change is in a more-conservative direction. All applicable UFSAR Chapter 15 transient acceptance criteria are met with the proposed amendment. Therefore, operation of the facility in accordance with the proposed amendment will not involve a reduction in a margin of safety.

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

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North Carolina.

Attorney for licensee: Mr. Albert Carr, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Project Director: Herbert N. Berkow.

Duke Energy Corporation (DEC), et al., Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: May 27, 1997, as supplemented by letters dated March 9, March 20, April 20, May 27, June 3, June 24, and July 7, 1998.

Description of amendment request: The proposed amendments would revise the current Technical Specifications (TS) of each unit to conform with NUREG–1431, Revision 1, ''Standard Technical Specifications—Westinghouse Plants.'' The staff had previously issued a Notice of Consideration of Issuance of Amendments published in the **Federal Register** on July 15, 1997 (62 FR 37940) covering all the proposed changes that were indeed within the scope of NUREG–1431. The staff subsequently published additional Notices of Consideration of Issuance of Amendments and Proposed No Significant Hazards Determination on May 6, 1998 (63 FR 25107 and 63 FR 25108 (two notices)) and on May 20, 1998 (63 FR 27761) to cover DEC's March 9, March 20, April 20, and May 27, 1998, supplements, which proposed changes that are beyond the scope of NUREG–1431.

On June 24, 1998, DEC identified additional beyond-scope changes. The following descriptions and proposed no significant hazard analyses cover only those beyond-scope changes. Associated with each change are administrative/editorial changes such that the new or revised requirements would fit into the format of NUREG–1431.

1. Current TS 4.8.1.1.2.f specifies that the fuel for the emergency diesel generators (EDGs) be periodically sampled for particulate contamination in accordance with ASTM–D2276–78. DEC proposed to relax this requirement, adopting instead the guidance of ASTM–D2276, Method A. The revised requirement would show up as TS 5.5.13.c of the Improved TS. No changes to the design and functions of the EDGs are proposed.

2. DEC proposed to change the required action due to inoperable channels of the containment pressure control system as currently contained in Table 3.3–3, Item 7. The revised requirement would show up as Action Item 16b in Table 3.3.2–1 of the Improved TS. No changes to the design and functions of the containment pressure control system are involved.

3. DEC proposed to revise current TS Table 4.3–1, Functions 16 and 17. The revised requirements, to show up as Table 3.3.1–1 Functions 15 and 16.b, would add an actuation logic test surveillance for the reactor trip system interlocks and the safety injection input from the engineered safety feature actuation system. No changes to the design and functions of these systems are involved.

Basis for proposed no significant hazards consideration determination:

As required by 10 CFR 50.91(a), DEC has provided its analyses of the issue of no significant hazards consideration for each of the above proposed changes. The NRC staff has reviewed DEC's analyses against the standards of 10 CFR 50.92(c). The NRC staff's analysis is presented below.

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

For all the changes the answer is ''no.'' The proposed changes will not affect the safety function of the subject systems. There will be no direct effect on the design or operation of any plant structures, systems, or components. No previously analyzed accidents were initiated by the functions of these systems, and the systems will continue to perform their functions in mitigating consequences of previously analyzed accidents. Therefore, the proposed changes will have no impact on the consequences or probabilities of any previously evaluated accidents.

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

For all the changes the answer is ''no.'' The proposed changes would not lead to any hardware or operating procedure change. Hence, no new equipment failure modes or accidents from those previously evaluated will be created.

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

For all the changes the answer is ''no.'' Margin of safety is associated with confidence in the design and operation of the plant. The proposed changes to the TS do not involve any change to plant design, operation, or analysis. Thus, the margin of safety previously analyzed and evaluated is maintained.

Based on this analysis, it appears that the three standards of 10 CFR 50.92(c) are satisfied for each of the proposed changes. 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: Mr. Paul R. Newton, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Project Director: Herbert N. Berkow.

Duke Energy Corporation, Docket No. 50–287, Oconee Nuclear Station, Unit 3, Oconee County, South Carolina

Date of amendment request: July 16, 1998.

Description of amendment request: The proposed change would extend, on a one-time basis, certain specified Technical Specification surveillances that are required to be performed at a frequency of 18 months from the maximum allowed frequency of 22 months, 15 days, to a maximum of 24 months. The following surveillances are involved: (a) Standby Shutdown Facility (SSF) Reactor Coolant System (RCS) Pressure Instrument Calibration; (b) SSF RCS Pressurizer Level Instrument Calibration; (c) SSF RCS Makeup Pump Flow Instrument Calibration; (d) Reactor Protective System (RPS) RCS Flow Instrument Calibration; (e) RPS RCS Pressure Instrument Calibration; and (f) Low Pressure Injection System Pump Discharge Valves LP–17 and LP–18 Manual Cycle.

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

This proposed change has been evaluated against the standards in 10 CFR 50.92 and has been determined to involve no significant hazards, in 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?

No. A review of the previous two instrument channel tests and calibrations, and two manual valve cycle tests discussed in this amendment request concluded that no adverse effects should occur as a result of the one-time extension.

There is a high level of confidence that the instruments and valves should be available to perform their intended function during the requested extension period. Thus, the probability and consequences of an accident previously evaluated will not be significantly increased.

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

No. Since the one-time extension should not cause any adverse effects on Standby Shutdown Facility, Reactor Protective System or the Low Pressure Injection system, a new or different kind of accident from the accidents which were previously evaluated will not occur. The Standby Shutdown Facility, Reactor Protective System or the Low Pressure Injection system should be available to perform their intended function during the requested extension period.

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

No. The margin of safety will not be significantly reduced by this amendment request because the Standby Shutdown Facility, Reactor Protective System or the Low Pressure Injection system should be available to perform their intended function during the requested extension period. In addition, the review of the previous tests and calibrations which are discussed in the amendment request concluded that no adverse effects should occur as a result of the one-time extension.

Duke [Energy Corporation] has concluded, based on the above information, that there are no significant hazards involved in this amendment request.

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: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina.

Attorney for licensee: J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW., Washington, DC.

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: April 28, 1998.

Description of amendment request: The proposed amendment would change the scope and frequency of volumetric and surface inspections for the reactor coolant pump motor flywheels. The current prescribed frequency and scope are contained in U.S. NRC Regulatory Guide 1.14, Regulatory Positions C.4.b.1 and C.4.b.2. The proposed revision reflects the frequency and scope of volumetric and surface examinations, which has been reviewed and approved by the NRC, as stated in the Safety Evaluation for Topical Report WCAP–14535A.

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

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

The CR–3 [Crystal River Unit 3] components addressed by this proposed change are the Reactor Coolant Pumps (RCPs), identified by plant tagging procedures as RCP–1A, RCP–1B, RCP–1C, and RCP–1D. The RCPs are vertical, single stage, single suction, shaft seal, centrifugal pumps. The RCPs ensure that adequate cooling water is circulated through the reactor coolant system. Following loss of power to the RCP motor, the flywheel, in conjunction with the impeller and motor rotating assembly, provide sufficient rotational inertia to assure adequate coolant flow during RCP coastdown, thus providing adequate core cooling. The maximum loading on the RCP motor flywheel results from overspeed following a large loss of coolant accident (LOCA). The estimated maximum speed in the event of a LOCA was established conservatively. The proposed change does not affect that analysis. Reduced coastdown times due to a single failed flywheel is bounded by the locked rotor analysis, therefore it will not place the plant in an unanalyzed condition.

Reducing the frequency of inspection, as proposed, will not significantly increase the probability of an accident previously evaluated. CR–3 is not specifically analyzed for a flywheel failure accident. The design, fabrication, and testing of the flywheels in accordance with the guidance found in Regulatory Guide 1.14 minimizes the potential for flywheel failure. Nevertheless, 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 flywheel. However, inspections are proposed to continue at a frequency of once every ten years as a conservative measure. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The purpose of the RCP motor flywheel inspection is to identify flaws that could lead to failure of the flywheel. The design, fabrication, and testing of the flywheels in accordance with the guidance found in Regulatory Guide 1.14 minimizes the potential for flywheel failure. No new failure mode is introduced due to the change in flywheel inspection frequency since the proposed changes do not involve the addition or modification of equipment, nor alter the design or operation of affected plant systems, structures or components. Therefore, these changes do not create a possibility of a new or different kind of accident from any previously evaluated.

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

As shown in the topical report, 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 excellent 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. The non-destructive examination acceptance criteria is not changing as a result of the proposed LAR. Thus, the margin of safety is not reduced significantly by the proposed change in inspection frequency.

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

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

Attorney for licensee: R. Alexander Glenn, General Counsel, Florida Power Corporation, MAC—A5A, P. O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: May 27, 1998.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TS) to remove the requirement for safety injection tanks (SITs) to be operable in reactor operational Mode 4.

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

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

The proposed amendment does not involve changes to previously evaluated accident initiators. The proposed TS changes related to removal of the requirement for safety injection tanks to be operable in MODE 4 do not impact the results of existing accident analyses, and have no adverse impact on any plant system performance.

The function of each SIT is to provide early reactor core reflood in the event of a LBLOCA [large break loss-of-coolant accident]. Safety injection tanks are not required for mitigating the consequences of large RCS pipe ruptures in MODE 4, and the proposed change to TS 3.5.1 will delete the requirement for SIT operability when in this mode. Due to the reduced initial stored energy and decay heat generation rate consistent with operation in the shutdown modes, the required operable HPSI [high-pressure safety injection] pump is sufficient to perform the function of reactor vessel reflood and coolant inventory make-up. Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendment will not change the physical plant or the modes of operation defined in the facility license. The changes do not involve the addition of new equipment or the modification of existing equipment, nor do they alter the design of St. Lucie plant systems described in the Updated Final Safety Analysis Report (UFSAR). There are no adverse effects on any system performance due to the proposed TS changes, and the plant configuration will continue to remain consistent with assumptions used in the existing accident analyses. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed TS changes have been evaluated with respect to the applicable safety analyses. FPL [Florida Power and Light Co.] determined from this new evaluation that safety injection tanks are not required to prevent core uncovery during a loss of coolant accident initiated in MODE 4. Due to the reduced core heat removal requirements in this lower mode and in the absence of substantial core uncovery, fuel cladding temperatures and clad oxidation will remain at low levels, long term cooling will be maintained, and 10 CFR 50.46 acceptance criteria will be satisfied. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Indian River Community College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34981–5596.

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: June 29, 1998.

Description of amendment request: This Technical Specification change request replaces in its entirety, a previously submitted request dated February 22, 1996, and published in the **Federal Register** on March 27, 1996 (61 FR 13525). This request greatly reduces the scope of the previous request. It retains the provision to delete the requirement that the biennial inspection of the Emergency Diesel Generators (EDGs) be performed during shutdown, permits skipping diesel starting battery capacity test for recently installed batteries, and increases the minimum loading during diesel testing from 20% to 80%. In addition, there are wording changes to enhance clarity, and a typographical error is corrected.

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. State the basis for the determination that the proposed activity will or will not increase the probability of occurrence or the consequences of an accident.

The proposed activity deletes the requirement to inspect EDGs during shut down from the Technical Specifications and permits skipping diesel starting battery capacity tests of recently installed batteries. The minimum loading during the testing of the diesels has been increased from 20% to 80%. In addition, wording changes were made to enhance clarity and a minor typographical error was corrected. During reactor operations other power sources are available to compensate for one diesel being out of service. The inspections and testing will continue to be done with the same intervals and the 80% loading is a more stringent requirement. Therefore, these changes do not affect the design or performance of the EDGs or their ability to perform their design function.

2. State the basis for the determination that the activity does or does not create a possibility of an accident or malfunction of a different type than any previously identified in the [safety analysis report] SAR.

The EDGs are not the source of any accident described in the SAR. These changes do not modify the design or performance of the EDGs and do not affect plant functions or actions. Current specifications permit one diesel generator to be inoperable for up to 7 days and this change will not impact that time frame. Therefore, the proposed change does not create the possibility of an accident or malfunction of a different type than those previously identified.

3. State the basis for the determination that the margin of safety is not reduced.

The proposed changes are designed to improve EDG reliability and availability during shutdown periods by providing flexibility in the scheduling and performance of maintenance. The surveillance intervals are unchanged and operability requirements are not modified. The proposed activity does not alter the basis of any technical specification that is related to the establishment or maintenance of a nuclear safety margin. Therefore, the margin of safety is not significantly reduced by this action.

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: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

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

NRC Project Director: Cecil O. Thomas.

Niagara Mohawk Power Corporation, Docket No. 50–220, Nine Mile Point Nuclear Station, Unit 1 (NMP1), Oswego County, New York

Date of amendment request: June 19, 1998.

Description of amendment request: The proposed amendment would update Technical Specification (TS) 3.2.2, ''Minimum Reactor Vessel Temperature for Pressurization,'' and the associated TS Bases pages. TS 3.2.2 contains tables and figures that limit the minimum reactor vessel temperature for a given pressure. The limits are based upon the number of Effective Full Power Years (EFPY) of core operation. The current tables and figures are valid for up to 18 EFPYs of core operation. The proposed amendment will substitute new tables and figures that are valid for 20, 24 and 28 EFPYs. The word ''leakage'' would be added to clarify that this TS applies to both leakage and hydrostatic tests.

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 operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The changes to the P–T [pressure and temperature] curves are being proposed to preclude brittle fracture of RPV [reactor pressure vessel] materials for up to 28 EFPYs. In addition to the leakage/hydrostatic test curve for 28 EFPYs, leakage/hydrostatic test curves have been prepared for exposures up to 20 EFPYs and up to 24 EFPYs to shorten outage time for startups conducted prior to these exposures. Safety margins specified in 10 CFR Part 50, Appendix G and Appendix G to Section III of the ASME [American Society of Mechanical Engineers] Code will continue to be met for each of these curves. Also, the proposed changes do not affect the probability of any accident precursors. Therefore, operation in accordance with the proposed change will not involve a significant increase in the probability of an accident previously evaluated.

The RPV, as part of the reactor coolant system, provides a barrier to the release of reactor coolant and subsequent radiological consequences. Operation in accordance with the proposed amendment will preclude brittle fracture of the RPV consistent with current requirements, and consequently, not affect the consequences of any accidents. Therefore, operation of NMP1 [Nine Mile Point Unit 1] in accordance with the proposed amendment will not involve a significant increase in the consequences of an accident previously evaluated.

2. The operation of Nine Mile Point Unit 1, 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 proposed change does not involve any physical alterations to plant configurations or introduce any new accident precursors which could initiate a new or different kind of accident. The proposed change does not affect the intended function of the RPV nor does it affect the operation of the RPV in a way which would create a new or different kind of accident. The changes to the P–T curves are being proposed to preclude brittle fracture of RPV materials for up to 28 EFPYs. Safety margins specified in 10 CFR Part 50, Appendix G and Appendix G to Section III of the ASME Code will continue to be met. Therefore, operation of NMPI in accordance with the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The existing NMP1 P–T curves were developed using safety margins for brittle fracture found in 10 CFR PART 50 Appendix G and Appendix G to Section III of the ASME Code. The proposed NMP1 P–T operation curves, which are valid for up to 28 EFPYs of operation, were also developed using the safety margins for brittle fracture found in 10 CFR PART 50, Appendix G and Appendix G to Section III of the ASME Code. Accordingly, operation of NMPI in accordance with the revised P–T operating limits will continue to preclude brittle fracture of the RPV materials during plant heatup, cooldown, and leakage/hydrostatic test conditions with the same margin of safety that currently exists. Therefore, operation of NMP1 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 Project Director: S. Singh Bajwa.

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

Date of amendment request: May 22, 1997, as supplemented by letters dated June 12, 1997, August 28, 1997 and January 29, 1998.

Description of amendment request: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant Unit Nos. 1 and 2 TS 3/4.7.3.1, ''Plant Systems—Vital Component Cooling Water System,'' to add new action statements and surveillance requirements for the component cooling water (CCW) surge tank pressurization system. CCW surge tank pressurization system requirements currently exist in an equipment control guideline, but are proposed for inclusion in TS because the CCW surge tank pressurization system is required to support licensing basis assumptions for a design basis loss-of-coolant accident.

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 component cooling water (CCW) surge tank pressurization system is designed to mitigate the consequences of an accident, and cannot initiate an accident.

The proposed changes to the Technical Specifications (TS) incorporate requirements for the CCW surge tank pressurization system to assure that the consequences of an accident are not increased. The CCW surge tank pressurization system was installed to restore the component cooling water system to its original design and licensing basis. The design of the CCW surge tank pressurization system ensures that a minimum pressure of 17 psig is maintained in the surge tank at the initiation of a design basis loss of coolant accident. This minimum pressure is sufficient to ensure that boiling will not occur in the containment fan cooler units (CFCUs), assuming the worst case accident conditions with a concurrent loss of offsite power (LOOP).

Therefore, the addition of these new requirements does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The CCW surge tank pressurization system is designed to mitigate the consequences of an accident, and cannot initiate an accident.

The proposed TS changes incorporate requirements for the CCW surge tank pressurization system. Installation of the CCW surge tank pressurization system provides assurance that boiling in the CFCUs will not occur, assuming the worst case accident, with a concurrent LOOP.

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

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

The proposed changes to the TS incorporate requirements for the CCW surge tank pressurization system to assure that the consequences of an accident are not increased. The design of the CCW surge tank pressurization system ensures that a minimum pressure of 17 psig is maintained in the surge tank at the initiation of a design basis accident. The minimum pressure is sufficient to ensure that boiling will not occur in the CFCUs, assuming the worst case accident conditions with a concurrent LOOP.

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

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

Local Public Document Room Location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

Attorney for Licensee: Christopher J. Warner, Esq., Pacific Gas & Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Project Director: William H. Bateman.

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

Date of application for amendments: March 20, 1998, as revised by letter dated June 26, 1998.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to permit incorporation of an End-of-Cycle Recirculation Pump Trip.(EOC–RPT) System at Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3.

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

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

The addition of the EOC–RPT System will not involve a significant increase in the probability or consequences of an accident previously evaluated. The EOC–RPT System has been designed to appropriate standards and specifications to ensure that the ability of the plant to mitigate the effects of accidents is maintained. Each division is electrically, mechanically, and physically independent to meet the single failure criterion.

The EOC–RPT System will improve the reactor core thermal response following a turbine trip transient caused by either a turbine control valve fast closure or a turbine stop valve closure. The EOC–RPT will be relied upon to reduce the fuel thermal mechanical transient excursion such that fuel thermal limits are not violated. Under conditions when the system is inoperable, more conservative thermal limits will be enforced.

The new system will utilize existing RPS [Reactor Protection System] logic to initiate the Reactor Recirculation System (RRS) pump trips on a turbine generator trip and a generator load rejection event. The inputs to RPS used by EOC–RPT will be from turbine stop valve (TSV) limit switches and turbine control valve (TCV) oil pressure switches. There will be no direct interface between the EOC–RPT System and the main turbine control system. Thus the new system can not initiate a turbine trip or generator load rejection event. This change does not result in significant increase in the probability of events described in the UFSAR [Updated Final Safety Analysis Report]. Additionally, the probability of inadvertent single or dual recirculation pump trips due to the addition of the EOC–RPT components will not be significantly increased by this modification.

No new challenges to the reactor coolant pressure boundary will result from the incorporation of the EOC–RPT System which could result in a significant increase in the consequences of an accident.

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

The ECO-RPT System has been designed to appropriate standards and specifications to ensure that no new sequence of events or failure modes will occur such that a transient event will escalate into a new or different type of accident.

The PBAPS UFSAR evaluates several recirculation pump trip events, including the limiting case of a pump seizure. A spurious dual EOC–RPT pump trip is similar to other RRS pump trip events evaluated in the UFSAR and does not represent a different type of accident.

Additionally, this modification will not create any new failure mode or sequences of events that could lead to a different type of accident than previously evaluated. The new EOC–RPT System will not involve any new challenges to a fission product barrier. The EOC–RPT System does not make any changes to the design function of the RRS. Therefore, the new equipment installed by this modification cannot create the possibility of a different type than previously evaluated in the SAR [Safety Analysis Report].

The EOC–RPT System is classified as important-to-safety. Failure or malfunction of the new equipment will not prevent or affect the ability of safety-related or important-to-safety systems to respond to the design basis accidents described in the FSAR [Final Safety Analysis Report].

There will be no software used in the EOC–RPT System. The system logic consists of two electrically and physically separated trip systems; one will be used to trip one EOC–RPT System breaker, and the other will be used to trip the second EOC–RPT System breaker for each pump.

The design of this modification assures that the new system is not susceptible to electromagnetic (EM) emissions and will not cause inadvertent operation of existing plant equipment due to EM emissions.

Based on the previous discussion, the possibility of a new or different kind of accident from any accident previously evaluated will not be created.

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

There are no significant reductions in any margin of safety previously approved by the USNRC as a result of this change to the TS. The EOC–RPT System will ensure that fuel thermal limits are not exceeded during the limiting transient. In the event that the EOC–RPT System is determined to be inoperable, specific operating limits are provided in the COLR. In all cases, thermal limits are not exceeded and the margin of safety is not significantly reduced.

The plant LOCA response will not change for present core configurations (i.e., 9 x 9 fuel) with the EOC–RPT System installed. For GE 8 x 8 fuel, which could be used at a future time, there could be a small increase in Peak Cladding Temperature (PCT). This increase would still be well below the 2200° F acceptance limit defined in 10 CFR 50.46.

There will be no significant reduction in the margin of safety as previously approved by the USNRC, since the calculated increase in peak cladding temperature for a core containing limit 8x8 fuel design (BP/P8 x 8R) is a small increase above the previously analyzed peak cladding temperature. Additionally, this modification does not impact the safety function of the RRS piping, thus reactor coolant pressure boundary safety limits are not affected.

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: Robert A. Capra.

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, PeachBottom Atomic Power Station, Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: May 1, 1998

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to delete requirements for the functional testing of the safety relief valves (SRVs) during each unit startup at Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3.

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

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

The proposed Technical Specification changes to the requirement for functional testing of the SRVs during each unit startup will not significantly increase the probability or consequences of an accident previously evaluated. Elimination of the functional test will not prevent the SRVs from performing their intended safety function. The proposed change to delete the SRV functional test at power should delete a potential initiator of SRV leakage. The remaining testing and inspections will continue to adequately demonstrate the operability of the SRVs for both the safety and depressurization modes.

As a result of deleting the requirement for functional testing of the SRVs during each unit startup and replacing these requirements with the proposed tests contained TS SR [surveillance requirement] 3.4.3.2 and 3.5.1.12, the only change in the frequency of testing of the SRV components is that the main valve disc of the SRVs will be tested every two cycles (approximately four years) as compared to the current one cycle (approximately two years) frequency. As described above, the lift test of the main valve disc is currently performed at an offsite facility. A review of offsite testing data for the years 1987 through 1998 was performed for the PBAPS, Units 2 and 3 SRVs. Since the design of the SRVs is to ensure operation of the overpressurization protection and the ADS [Automatic Depressurization System] function is to reduce reactor pressure during a small break LOCA [loss-of-coolant accident], the review consisted of looking for any failures of the main valve disc to stroke open during setpoint actuation. This review consisted of reviewing ''as-found'' test data since any failures following a rebuild would be found during the final certification testing. Based on a review of as-found data, it was concluded that there were no reported cases of the main disc failing to open during setpoint pressure testing. Therefore, deleting the requirement for functional testing of the SRVs during each unit startup is not expected to negatively impact these test results.

Therefore, eliminating the functional test is not expected to negatively impact these test results or involve a significant increase in the probability of an accident previously evaluated.

As discussed in the PBAPS, Units 2 and 3 Updated Final Safety Analyses Report (UFSAR), analyzed events resulting in a nuclear system pressure increase, such as MSIV [main steam line isolation valve] closure, generator load rejection, turbine trip, failure of the turbine bypass valves to open, and loss of main condenser vacuum, take credit for the SRVs opening to mitigate the consequences of these events. The proposed changes will not increase the consequences of these events, since a series of remaining tests will ensure all SRV components will function. The SRVs will therefore be capable of performing their design functions.

SRV second stage valve leakage can be increased as a result of corrosion/debris introduced on the seating area surface. Second stage leakage, if allowed to continually increase, will eventually start to depressurize the volume above the SRV main valve piston to the extent that sufficient differential pressure will lift the main valve disc. Reactor vessel coolant inventory decrease due to an inadvertent opening of a Safety Relief Valve is an abnormal operating transient event. This event can be a precursor to fuel failure due to gradual loss of coolant, and the mitigation is similar to the small break LOCA. Under the proposed change, it is expected that the probability of SRV leakage will decrease, thus the probability of occurrences of an inadvertent SRV actuation is reduced, therefore reducing the probability or consequences of an accident previously evaluated.

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

The SRVs will not be operated or tested in a manner contrary to their design. As a result, no new mode of operation is introduced. Therefore, the revised testing will not create a new failure mode of the SRVs which could create the possibility of a new or different kind of accident from any previously evaluated. Since other tests, taken together, confirm the entire SRV assembly functions adequately, this proposed change is justified. The proposed change to delete the SRV functional test at power will not impact the ability of the SRV to open and provide their intended safety function.

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

By removing the Technical Specification requirements to perform the in-situ functional testing during startup, the probability of inadvertently opening of a SRV should be reduced through the elimination of a potential initiator of SRV second stage disc leakage and subsequent erosion. This Technical Specification change will aid in decreasing SRV leakage and improve SRV reliability at power operations. Eliminating the SRV in-situ functional test during startup will increase the margin of safety during operations, transients, or accidents. Remaining surveillance testing and inspections assure each component necessary for successful opening of the SRV function properly as designed.

Removal of the functional test will not negatively impact the Technical Specifications lift setpoints of the SRVs necessary for the function of the safety mode. The functional test does not completely test the safety mode of the SRV which is based on the Technical Specifications lift setpoints.

Offsite testing at operating steam pressure ensures the operability of the SRV pilot, second stage, and main valve function. The valves are refurbished and post maintenance testing is performed at a steam pressure of 1040 psig. Upon successful test completion, the valve receives written certification from the lab and is returned to PBAPS for reinstallation. To receive certification, the valve must have zero main seat leakage and meet the acceptance criteria for setpoint pressure. These tests satisfy the requirements of the PBAPS IST [Inservice Testing] Program and TS. The tests contained in the proposed TS SR 3.4.3.2 and 3.5.1.12 will verify the operation of the solenoid and second stage disc movement of all 11 SRVs in the depressurization mode.

The remaining segments of the SRV tests verify the ability of the SRV logic. In summary, this change will not involve a significant reduction in the margin of safety, because of the reduction in SRV degradation, and the remaining tests confirm the valves will function properly when required.

The NRC staff has reviewed the licensee's analysis and, based on this review, this 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: Robert A. Capra.

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

Date of amendment request: June 16, 1998.

Description of amendment request: The proposed change would relocate the Safety Review Committee (SRC) review, audit, and related record keeping requirements from the Technical Specifications (TSs) to Chapter 17 of the Final Safety Analysis Report (FSAR).

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

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously analyzed?

Response: This amendment application does not involve a significant increase in the probability or consequences of an accident previously analyzed. The relocation of the SRC review, audit, and related record keeping requirements from the TS to the FSAR does not alter the performance or frequency of these activities. Future changes to the QA [Quality Assurance] program, located in Chapter 17 of the FSAR, which constitute a reduction in commitments, are governed by 10 CFR 50.54(a). Therefore, sufficient controls for these requirements exist and these changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

Response: This amendment application does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes involve the relocation of SRC requirements from the TS to the FSAR. Relocation of these requirements does not affect plant equipment or the way the plant operates. The reviews, audits, and record keeping will continue to be performed in the identical manner as they are currently being performed. Therefore, the proposed revisions cannot create a new or different kind of accident.

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

Response: This amendment application does not involve a significant reduction in a margin of safety. The requested Technical Specification revisions relocate SRC review, audit and related record keeping requirements from the TS to the FSAR. These requirements are not being altered by this relocation. The reviews, audits, and record keeping will continue to be performed in the identical manner as they are currently being performed. Any changes to these requirements which constitute a reduction in commitments will be processed in accordance with 10 CFR 50.54(a). Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

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

NRC Project Director: S. Singh Bajwa, Director.

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

Date of amendment request: July 10, 1998.

Description of amendment request: The proposed changes would relocate portions of reactor coolant chemistry requirements from the technical specifications (TSs) to licensee-controlled procedures. Changes to the relocated requirements will then be controlled by the provisions of 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 TS amendment will not significantly increase the probability or consequences of any previously evaluated accidents.

The proposed changes simplify the TS, meet regulatory requirements for relocated TS, and implement the recommendations of the Commission's Final Policy Statement on TS improvements. Future changes to these requirements will be controlled by 10 CFR

50.59. The proposed changes are administrative in nature and do not involve any modification to any plant equipment or affect plant operation. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of any previously evaluated accident.

2. The proposed TS amendment will not create the possibility of a new or different kind of accident.

The proposed changes are administrative in nature, do not involve any physical alterations to any plant equipment, and cause no change in the method by which any safety related system performs its function. Therefore, this proposed TS amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes are administrative in nature, will not alter the basic regulatory requirements, and do not affect any safety analyses. Therefore, no margin of safety is reduced as a result of these changes.

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

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

Attorney for licensee: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Project Director: S. Singh Bajwa, Director.

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

Date of amendment request: June 5, 1997, and supplemented April 21, 1998.

Description of amendment request:

Part 1—DG Online Testing:

The proposed amendment involves the testing of the standby diesel generators (DGs) and revises the Watts Bar Unit 1 (WBN) Technical Specifications (TSs) to allow additional testing of the DGs on-line during MODES 1 and 2. The proposed changes affect Surveillance Requirement (SR) 3.8.1.14. The testing performed for this surveillance fulfills the requirements of Regulatory Guide 1.9, ''Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants.'' This testing is performed once every 18 months to ensure that the DGs can start and run continuously for an interval of not less than 24 hours. Specifically, the proposed amendment revises SR 3.8.1.14 and its associated Bases to delete the note which prohibits the performance of the on-line 24 hour test during MODES 1 or 2.

Part 2—DG Battery Testing:

As currently written, the TSs permit testing of the DG batteries and chargers only during MODES 5 and 6 when operability of all four DGs is not required. The proposed amendment would revise the Watts Bar Unit 1 TSs to allow testing of the DG batteries and battery chargers during MODES 1, 2, 3, and 4 as well. Implementation of these changes will require entry into Action B.4 of TS 3.8.1 for the affected diesel.

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:

Part 1—DG Online Testing

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

The proposed amendment to allow the 24-hour DG endurance run to be conducted during any mode of operation does not significantly increase the probability or consequences of an accident previously evaluated in Chapter 15 of the FSAR [Final Safety Analysis Report] since the capability to safely shutdown the plant following a LOOP [loss of offsite power], LOCA [loss-of-coolant accident] or LOCA/LOOP coincident with a single failure is maintained throughout the surveillance test. The 24-hour endurance test does not disable any of the automatic actuations and interlocks of the DG control functions, nor prevent the satisfactory completion of the LOOP or LOCA/LOOP loading sequence if a LOOP or LOCA signal is received at any time during the test. Required Class-1E onsite power operability during normal operation, shutdown cooling, loss of offsite power, and accident conditions will be the same.

In addition, the performance of proposed Surveillance Requirement 3.8.1.14 during MODES 1 or 2 will not significantly increase the consequences of perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems. Performance of proposed Surveillance Requirement 3.8.1.14 during MODES 1 or 2, or failure of the surveillance, will not cause, or result in, an anticipated operational occurrence with attendant challenges to plant safety systems that has not been previously analyzed for the existing monthly surveillances.

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

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

The requested changes do not result in a new or different kind of accident from that previously analyzed in WBN's Final Safety Analysis Report. The changes propose to eliminate restrictions of the plant operating modes in which standby DG system testing may be performed but does not change the type of testing performed and are not due to modification of the system design. NRC's assessment of the testing of the DGs in the configuration proposed is documented in Section 8.3.1.12 of Supplements 13 and 14 of the Safety Evaluation Report and in letters dated June 20, 1991, and March 28, 1994.

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

As previously stated, performance of proposed Surveillance Requirement 3.8.1.14 during Modes 1 or 2 will not cause, or result in, an anticipated operational occurrence with attendant challenges to plant safety systems that has not been previously analyzed for the existing monthly surveillances. Therefore, implementation of the proposed amendment will not reduce the margin of safety for this system.

Part 2—DG Battery Testing

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

The proposed changes to the TSs apply only to the DG battery system and do not in any way affect the vital battery system or safety system loads supplied by the vital battery system. The changes do not result in a condition where the design or function of the DGs or DG battery systems would be modified. The DG battery subsystems supply only the control and field flashing power to support a single DG and do not supply any other unrelated system loads or functions. Therefore, manipulation of the DG battery system is not a credible means of perturbing the vital power distribution system and challenging safety systems. In addition, the surveillances for the DG batteries are required to be performed only once every 18 months.

A DG declared inoperable due to the testing must be returned to operable status within 72 hours in accordance with Action B.4 of TS 3.8.1. To ensure this could be achieved, the results of previous performances of the SRs were reviewed. From this review, it was established that in accordance with LCO [limiting condition for operation] 3.8.6, Table 3.8.6–1, Note c, the batteries can be restored within 72 hours to a condition where the charging current is less than 1 ampere. Achieving this charging current for the DG batteries is acceptable for meeting specific gravity limits following a battery recharge for a maximum of 31 days. In addition, the DG sets are occasionally removed from the standby condition to perform preventative and/or corrective maintenance. The intent is to perform this testing in conjunction with other required maintenance activities such that adverse effects on diesel unavailability are minimized. Compliance with the 10 CFR 50.65 Maintenance Rule program requirements for diesel unavailability ensures that any diesel inoperability incurred by this change is minimized.

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

The requested changes do not result in a new or different kind of accident from that previously analyzed in WBN's Final Safety Analysis Report. The changes propose to eliminate restrictions of the plant operating modes in which DG battery system testing may be performed but does not change the type of testing performed and are not due to modification of the system design. The requested changes will result in a DG being declared inoperable in accordance with Action B.4 of TS 3.8.1 for the duration of the testing, but does not impact the existing time limitations for the LCO. This change does not alter system performance and does not introduce any new accident initiators or scenarios.

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

The proposed amendment concerns only the conduct of testing but does not in any way affect the performance parameters of the safety system or in any way affect the ability of the system to perform its safety function of providing control and field flashing power for the DGs. Consequently, operation of the facility in accordance with the requested changes would not involve a significant reduction in the margin of safety.

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

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902.

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: June 26, 1998

Description of amendment request: The proposed amendment would revise the Watts Bar Nuclear Plant (WBN) Technical Specifications (TS) and associated Bases to delete the power range neutron flux high negative rate reactor trip function based on the analysis provided in Westinghouse Electric Corporation topical report WCAP–11394–P–A, ''Methodology for the Analysis of the Dropped Rod Event.''

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

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

The negative flux rate trip deletion does not increase the probability or consequences of core damage accidents resulting from dropped RCCA [rod cluster control assembly] events previously analyzed. The safety functions of other safety related systems and components, which are related to accident mitigation, have not been altered. All other primary protection (reactor trip and ESF) functions are not impacted by the elimination of the negative flux rate trip function. The consequences of accidents previously evaluated in the FSAR [final safety analysis report] are unaffected by this proposed change because no change to any equipment response or accident mitigation scenario has resulted. There are no additional challenges to fission product barrier integrity. No new radiological analyses are required. Therefore the proposed change will have no effect on the probability or consequences of accidents previously evaluated.

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

The negative flux rate trip deletion does not create the possibility of a new or different kind of accident than any accident already evaluated in the FSAR. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of this proposed change. The proposed modification does not challenge the performance or integrity of any safety-related systems.

It has been demonstrated that the function of the negative flux rate trip can be eliminated by the approved methodology described in WCAP 11394–P–A. A Watts bar specific analysis has confirmed that for the dropped RCCA and dropped RCCA bank event, no direct reactor trip or automatic power reduction is required to meet the DNB [departure from nucleate boiling] licensing basis for this Condition II event. The negative flux rate trip function is not credited as a backup for any other Chapter 15 event. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The margin of safety associated with the acceptance criteria for any postulated WBN accident is unchanged. It has been demonstrated that the function of the negative flux rate trip can be eliminated by the approved methodology described in WCAP 11394–P–A. Watts Bar specific analysis has confirmed that the dropped RCCA and dropped RCCA bank acceptance criteria (DNB) continue to be met. Conformance to the regulatory criteria for plant operation with the negative flux rate trip deletion is demonstrated, and regulatory limits (DNB) are not exceeded. The modification will have no effect on the availability, operability, or performance of the safety-related systems and components. Therefore, the proposed license amendment does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902.

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: June 30, 1998.

Description of amendment request: The proposed license amendment revises Technical Specification 3.1.7, ''Standby Liquid Control System.'' The purpose of the proposed change is to increase the boron concentration in the Standby Liquid Control System for the Perry Nuclear Power Plant Cycle 8 fuel design, and to provide margin for future cycles as 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:

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

The proposed change will not significantly increase the probability or consequences of an accident previously evaluated. The change will only vary the ratio of borax to boric acid that resides within the Standby Liquid

Control System (SLCS) as the neutron absorber.

Changing the definition of the solution from a mixture of Sodium Pentaborate having a molar ratio of 0.200, to a mixture of borax and boric acid having a nominal molar ratio of 0.229, does not degrade the stability of the solution, change the mixing accuracy requirements, or reduce the temperature margins that might add to the risk of solution crystallization. For each cycle, the reload safety analysis confirms that the SLCS boron concentration will satisfy the Technical Specification requirements for the Perry Nuclear Power Plant (PNPP).

The 5°F margin of safety for solution solubility will continue to be maintained and supported by the Containment Building ambient temperatures and additionally supplemented by auto initiated heating on the SLCS tank and piping. The chosen borax and boric acid molar ratio will continue to maintain a limiting chemical addition mass, to the plus 5°F solubility limit, greater than or equal to the current 0.200 mixture. Any inaccuracies associated with tank temperature, tank volume, chemical analysis, and initial and subsequent chemical additions to the tank will also remain the same.

The primary reactivity control system for postulated accident conditions is the control rod system. The SLCS is a redundant reactivity control system to the control rod system and is used in special plant capability demonstration events cited in Appendix A of the Updated Safety Analysis Report (USAR), Chapter 15, which are extremely low probability non-design basis postulated incidents. There are no postulated accidents evaluated in USAR Chapter 15 that take credit for two or more reactivity control systems preventing or mitigating each accident. There is no increase to the radiological consequences of postulated incidents with the proposed change.

With the implementation of this proposed change, the SLCS will continue to operate and perform to all of its current requirements for providing shutdown margin under operating and ATWS conditions per 10 CFR 50, Appendix A, General Design Criterion (GDC) 26 and 10 CFR 50.62. The proposed change will not alter the operation of any plant equipment assumed to function in response to an analyzed event or otherwise increase its failure probability. Therefore, the proposed 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.

The proposed change does not impact the operation of the SLCS or the function of any of the active components. No new system interactions are created by this change and any parameters or conditions that could contribute to the initiation of accidents different than those already evaluated in the USAR are not impacted. The change will only vary the ratio of borax to boric acid that resides within the SLCS as the neutron absorber. The proposed values for solution molar ratio and boron concentration ensure that solution temperature margins are maintained greater than or equal to the current required margin to prevent solution crystallization. As a result, no new failure modes are being introduced.

Changing the definition of the solution from Sodium Pentaborate having a nominal molar ratio of 0.200, to a mixture of borax and boric acid having a nominal molar ratio of 0.229 does not degrade the stability of the solution, change the mixing accuracy requirements, or reduce the temperature margins that might add to the risk of solution crystallization.

Sufficient margin will be maintained to allow for expected deviations in the molar ratio and boron weight as the result of variations in product composition, test measurement inaccuracies, and for chemical addition inaccuracies. The boron concentration required within the SLC system to meet the required shutdown margin, will continue to be determined for each fuel cycle as part of the reload safety analysis per Technical Specifications. The borax and boric acid concentration will remain controlled via the Technical Specification Surveillance Requirements and the associated administrative procedures, USAR text, and existing licensing commitments.

The SLC system will meet its design basis requirements for the weight of boron injected and for maintaining the required temperature margin for system operation. As the result of the proposed change to increase the minimum boron concentration, a new minimum required SLC pump flow rate was determined for compliance with the NRC ATWS Rule 10 CFR 50.62.

The proposed change meets current regulations, maintains the fundamental safety principles of plant design, and the associated margins of safety. With the implementation of the proposed change, the SLCS will continue to operate and perform to all of its current requirements for providing shutdown margin under operating and ATWS conditions per 10 CFR 50, Appendix A, GDC 26 and 10 CFR 50.62. As a result, no new failure modes are being introduced. There are no changes in the methods governing normal plant operations, nor are the methods used to respond to plant transients altered. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The required margin of safety for the SLCS solution ensures an adequate margin of solubility such that no precipitation will occur in the SLC storage tank. The current margin is provided by maintaining a minimum solution temperature that is no less than the saturation temperature corresponding to the concentration of the solution in the storage tank plus 5°F.

This 5°F provides the adequate margin for inaccuracies associated with tank temperature, tank volume, chemical analysis, and initial and subsequent chemical additions to the tank. The proposed change does not impact the inaccuracies associated with tank temperature, tank volume, chemical analysis, and initial and subsequent chemical additions to the tank. The new analytical design values for the molar ratio and boron concentration will continue to maintain the solution temperature margins in excess of the current minimum specified to prevent solution crystallization.

Ambient temperatures within the building that houses the SLC storage tank, the Containment Building, will maintain the solution temperature. Additionally, the solution temperature is maintained by the presence of auto initiated tank heaters and pipe heat tracing. The 5°F margin will be maintained with the new SLCS mole ratio and higher boron concentration with the existing instrument setpoints and administrative controls.

The proposed change maintains the same reactor shutdown margin for the next fuel cycle and does not reduce the margin of safety for any system parameter as defined in the bases for the Technical Specifications. The proposed change will not physically alter the SLCS's physical configuration or components or introduce new system interactions that could produce any parameters or conditions that could contribute to a reduction of safety for any other system or scenario. The change will only vary the ratio of borax to boric acid that resides within the SLCS as the neutron absorber.

Therefore, with the implementation of this proposed change, the SLCS will continue to operate and perform to all of its current requirements for providing shutdown margin under operating and ATWS conditions per 10 CFR 50, Appendix A, GDC 26 and 10 CFR 50.62 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 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: Perry Public Library, 3753 Main Street, Perry, OH 44081.

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

NRC Project Director: Ronald R. Bellamy (Acting).

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

Date of amendment request: June 30, 1998.

Description of amendment request: The licensee proposes to delete the calibration requirements for emergency core cooling actuation instrumentation—core spray (CS) subsystem and low pressure coolant injection (LPCI) system auxiliary power monitor since the relays operate from a switched input and functional testing is sufficient to demonstrate the relay pickup/dropout capability.

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

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

The proposed change does not involve a change to the plant design or operation. The Auxiliary Power Monitor logic relays installed are tested to fully demonstrate operability without performance of a calibration on the pickup voltage value. The design intent of the relays is to start LPCI and CS pumps as soon as possible without causing loss of the normal or emergency power supplies and within the time frames specified in the LOCA analysis of record. The proposed change does not affect any of the parameters or conditions that contribute to initiation of any accidents previously evaluated. Thus, the proposed change cannot increase the probability of an accident previously evaluated.

The proposed change does not involve a change in the operation of the relays controlling [Residual Heal Removal] RHR and CS Pump start with normal power available nor the initial RHR pump start on a LOCA with normal power not available or the time delay start of the remaining RHR or CS pumps with normal power not available. Failure of the relays to pickup would still result in the start sequence for normal power not available. The logic for both start sequences is verified independent of an instrument calibration and is consistent with the LOCA analysis and the EDG load analysis, therefore, the proposed change does not significantly increase the consequences of any accident previously evaluated.

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

This proposed change will not involve any physical changes to plant systems, structures or components (SSC), or the manner in which these SSCs are operated or maintained. The calibration requirement has previously been considered to be met by performance of the Simulated Automatic Actuation Test. Deletion of the calibration requirement will not affect the RHR or CS Pumps starting on a LOCA signal, with or without an [Loss of Normal Power] LNP. The operability of the Auxiliary Power Monitor relays will still be tested under the Functional test and Trip System Logic and Simulated Automatic Actuation tests at the frequencies specified. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This proposed change to delete the calibration requirement for the CS and LPCI Auxiliary Power Monitor functions will not change operation of the RHR or CS Pump start sequences on a LOCA signal, with or without normal power available. The instantaneous logic sequence relays and time delay relays will function to initiate RHR and CS Pump start as designed. RHR and CS Pump start times will remain within the LOCA Safety Evaluation of record. Operability of the relays and associated circuitry are still demonstrated by the Functional test and associated Trip System Logic and Simulated Automatic Actuation tests. Therefore, this change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

Attorney for licensee: Mr. David R. Lewis, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

NRC Project Director: Cecil O. Thomas.

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

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

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

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

Date of amendment request: June 26, 1998.

Brief Description of amendment: The proposed amendment would revise Technical Specification (TS) 3.7.8, ''Ultimate Heat Sink (UHS),'' to permit an 8-hour delay in UHS temperature restoration period prior to entering the plant shutdown required actions. Also, for the duration of the restoration, service water system (SWS) temperature will be monitored hourly, and should the temperature exceed 99 degrees F, the plant will enter TS 3.7.8 required action A.1, and be in MODE 3 within 6 hours.

Date of publication of individual notice in the **Federal Register**: July 8, 1998 (63 FR 36967).

Expiration date of individual notice: July 22, 1998, for comments; August 7, 1998, for hearings.

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

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: March 3, 1998, as supplemented by letters dated April 24 and May 7, 1998.

Description of amendment request: The proposed amendments would revise Figure 5.1–1 of the Technical Specifications (TS) to show the new location of the meteorological tower. The meteorological tower will be relocated to a new location to facilitate use of the current location as a construction site. The proposed TS change does not change the related TS Section 5.1.1.

Date of publication of individual notice in **Federal Register**: June 29, 1998 (63 FR 35293).

Expiration date of individual notice: July 29, 1998.

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North Carolina.

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.

Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan

Date of application for amendment: January 28, 1998 (NRC–98–0002).

Brief description of amendment: The amendment revises technical specification surveillance requirements 4.8.2.1.a.2, 4.8.2.1.b, and 4.8.2.1.c.4 to accommodate new limits associated with the design of the replacement Division II 130/260-volt dc battery.

Date of issuance: July 9, 1998.

Effective date: July 9, 1998, with full implementation prior to restart from the sixth refueling outage.

Amendment No.: 121.

Facility Operating License No. NPF–43: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: February 25, 1998 (63 FR 9597).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 9, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Monroe County Library System, Ellis Reference and Information Center, 3700 South Custer Road, Monroe, Michigan 48161.

Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan

Date of application for amendment: December 10, 1997 (NRC–97–0105), as supplemented January 28 and April 9, 1998.

Brief description of amendment: The amendment revises Technical Specification (TS) 2.2.1, ''Reactor Protection System Instrumentation Setpoints,'' TS 3.3.1, ''Reactor Protection System Instrumentation,'' TS 3.3.6, ''Control Rod Block Instrumentation,'' TS 3.4.1.1, ''Recirculation Loops,'' and the associated Bases to accommodate an upgrade of the power range neutron monitoring system. The amendment also revises the first page of Table 3.3.6–2 to correct a typographical error in the title.

NRC has also granted the request of Detroit Edison Company to withdraw a portion of its December 10, 1997, application. The proposed change would have revised TS Surveillance Requirement 4.3.1.3 and its associated Bases to indicate response time testing is performed only on applicable channels. However, following discussions with the NRC staff, the licensee withdrew the proposed change in a letter dated April 9, 1998 (NRC–98–0037). For further details with respect to this action, see the application for amendment dated December 10, 1997, as supplemented above, and the licensee's letter dated April 9, 1998, which withdrew this portion of the application for license amendment, and the staff's Safety Evaluation enclosed with the amendment. The above documents 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 room listed below.

Date of issuance: July 13, 1998.

Effective date: July 13, 1998, with full implementation prior to restart from the sixth refueling outage. Implementation of this amendment shall include preparation of Design Calculation DC–5721, Volume I, and performance of a human factors review for the installation of the plant modification as described in the licensee's application dated December 10, 1997, as supplemented January 28 and April 9, 1998, and as evaluated in the staff's safety evaluation attached to this amendment.

Amendment No.: 122.

Facility Operating License No. NPF–43: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: January 14, 1998 (63 FR 2279). The January 28 and April 9, 1998, letters provided clarifying information and updated TS pages that were within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards considerations determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 13, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Monroe County Library System, Ellis Reference and Information Center, 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: April 20, 1998.

Brief description of amendments: The amendments revise Tables 3.3–3 and 4.3–2 of the Technical Specifications of each unit, correcting the operation mode applicability of the control room area ventilation actuation logic and relays from ''All'' to ''1, 2, 3, 4.''

Date of issuance: July 9, 1998.

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

Amendment Nos.: 167—Unit 1; 159—Unit 2.

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

Date of initial notice in **Federal Register**: May 20, 1998 (63 FR 27761).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 9, 1998.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: October 2, 1996.

Brief description of amendments: The amendments revise the ANO–1&2 TSs by relocating selected TS requirements related to instrumentation from the TS to the Updated Final Safety Analysis Report. The NRC provided guidance to all holders of operating licenses or construction permits for nuclear power reactors on the proposed TS changes in Generic Letter 95–10, ''Relocation of

Selected Technical Specifications Requirements Related to Instrumentation,'' dated December 15, 1995.

Date of issuance: July 13, 1998.

Effective date: July 13, 1998, to be implemented within 30 days.

Amendment Nos.: 192 and 191.

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

Date of initial notice in **Federal Register**: January 15, 1997 (62 FR 2188).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 13, 1998.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: March 27, 1997, as supplemented by letters dated April 3, July 21, October 23, November 13, and December 12, 1997, January 21, January 29, March 23, May 1, May 19, and May 21, May 28, and June 12, 1998.

Brief description of amendment: The amendment changes Appendix A Technical Specification by increasing the Spent Fuel Pool storage capacity from 1088 to 2398 fuel assemblies and by increasing the maximum fuel enrichment from 4.9 w/o (weight percent) to 5.0 w/o U–235.

Date of issuance: July 10, 1998.

Effective date: July 10, 1998.

Amendment No.: 144.

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

Date of initial notice in **Federal Register**: December 2, 1997 (62 FR 63732).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 10, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 12, 1998.

Brief description of amendments: The amendments revised Turkey Point Units 3 and 4 Facility Operating Licenses and Technical Specifications to remove certain license conditions and oudated references, and to incorporate an organizational change.

Date of issuance: July 9, 1998.

Effective date: July 9, 1998.

Amendment Nos.: 198 and 192.

Facility Operating Licenses Nos. DPR–31 and DPR–41: Amendments revised Turkey Point Units 3 and 4 Facility Operating Licenses and Technical Specifications.

Date of initial notice in **Federal Register**: April 8, 1998 (67 FR 17225).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 9, 1998.

No significant hazards consideration comments received: No.

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

Niagara Mohawk Power Corporation, Docket Nos. 50–220 and 50–410, Nine Mile Point Nuclear Station Unit Nos. 1 and 2, Oswego County, New York

Date of applications for amendments: May 15, 1998 (two letters, one for each unit).

Brief description of amendment: The amendments change administrative sections of the Technical Specifications to reflect a restructuring of licensee's Nuclear Division upper management organization.

Date of issuance: July 7, 1998.

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

Amendment Nos.: 162 and 83.

Facility Operating License Nos. DPR–63 and NPF–69: Amendments revise the Technical Specifications.

Date of initial notice in **Federal Register**: June 2, 1998 (63 FR 30026).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 7, 1998.

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.

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

NRC Project Director: S. Singh Bajwa.

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

Date of application for amendments: September 28, 1995, and April 23, 1998.

Brief description of amendments: The amendments revise Technical Specification 3/4.8.1.2, ''Electrical Power Sources—Shutdown,'' by adding a note to surveillance requirement 4.8.1.2 that identifies those surveillances which are required to be performed during Modes 5 and 6 (cold shutdown and refueling, respectively).

Date of issuance: July 14, 1998.

Effective date: July 14, 1998.

Amendment Nos.: 212 and 192.

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

Date of initial notice in **Federal Register:** November 8, 1995 (60 FR 56369).

The April 23, 1998, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 14,1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: September 29, 1998, as supplemented February 6, 1998, April 17, 1998, and June 4, 1998.

Brief description of amendment: This amendment revises the allowable value and trip setpoint for the main steam isolation high steam flow input into limiting condition for operation.

Table 3.3.2–1, function 4.d.

Date of issuance: July 14, 1998.

Effective date: July 14, 1998.

Amendment No.: 71.

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

Date of initial notice in **Federal Register:** October 22, 1997 (62 FR 54876).

The February 6, 1998, April 17, 1998, and June 4, 1998, 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 July 14, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610.

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

Date of application for amendments: June 6, 1996, as supplemented September 26, 1997, January 23, 1998, and May 19, 1998 (TS–372).

Brief description of amendments: Changes to the technical specifications administrative controls related to quality assurance, and other administrative and editorial changes.

Date of issuance: July 9, 1998.

Effective date: July 9, 1998.

Amendment Nos.: 233, 252, and 211.

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

Date of initial notice in **Federal Register:** September 25, 1996 (61 FR 50346).

The supplemental letters dated September 26, 1997, January 23, and May 19, 1998 did not change the original no significant hazards determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 9, 1998.

No significant hazards consideration comments received: None.

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

Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of application for amendment: April 24, 1998.

Brief description of amendment: This amendment changed Technical Specification (TS) Section 3/4.3.1.1, ''Reactor Protection System Instrumentation,'' TS Section 3/4.3.2.1, ''Safety Features Actuation System Instrumentation,'' TS Section 3/4.3.2.2, ''Steam and Feedwater Rupture Control System Instrumentation,'' and the associated TS bases. The TS tables of response time limits were relocated to the Davis-Besse Technical Requirements Manual. Other changes in these TS sections were also made consistent with the relocation.

Date of issuance: July 7, 1998.

Effective date: July 7, 1998.

Amendment No.: 225.

Facility Operating License No. NPF–3: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 7, 1998.

No significant hazards consideration comments received: No.

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

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: April 24, 1996, as supplemented December 15, 1997, and June 22, 1998.

Brief description of amendments: These amendments revise Technical Specifications (TS) Section 15.7, ''Radiological Effluent Technical Specifications (RETS).'' Portions of the RETS are moved to licensee-controlled documents consistent with Nuclear Regulatory Commission guidance on TS improvements. Other sections of the TSs have also been revised consistent with the removal of portions of the RETS.

Date of issuance: July 13, 1998.

Effective date: July 13, 1998, with full implementation within 45 days. Implementation shall include relocation of certain Technical Specification requirements to licensee-controlled documents, as described in the licensee's application dated April 24, 1996, as supplemented by letter dated December 15, 1997, and June 22, 1998, and evaluated in the staff's safety evaluation attached to the amendments.

Amendment Nos.: 184 and 188.

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

Date of initial notice in **Federal Register:** June 5, 1996 (61 FR 28620) The December 15, 1997, and June 22, 1998, submittals provided additional clarifying information and updated TS pages. This information was within the scope of the original **Federal Register** notice and did not change the staff's initial no significant hazards considerations determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 13, 1998.

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 22nd day of July 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 98–20111 Filed 7–28–98; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request

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

Extension: Rule 17f–1(g)—SEC File No. 270–30—OMB Control No. 3235–0290

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.), the Securities and Exchange Commission (''Commission'') has submitted to the Office of Management and Budget a request for extension of the previously approved collection of information discussed below.

- Rule 17f–1(g) Requirements for reporting and inquiry with respect to missing, lost, counterfeit or stolen securities.

Rule 17f–1(g), under the Securities Exchange Act of 1934 (''Act''), requires that all reporting institutions (i.e., every national securities exchange, member thereof, registered securities association, broker, dealer, municipal securities dealer, registered transfer agent, registered clearing agency, participant therein, member of the Federal Reserve System and bank insured by the FDIC) maintain and preserve a number of documents related to their participation in the Lost and Stolen Securities Program (''Program'') under Rule 17f–1. The following documents must be kept in an easily accessible place for three years, according to paragraph (g): (a) copies or all reports of theft or loss (Form X–17F–1A) filed with the Commission's designee: (b) all agreements between reporting institutions regarding registration in the Program or other aspects of Rule 17f–1; and (c) all confirmations or other information received from the Commission or its designee as a result of inquiry.

Reporting institutions utilize these records and reports (a) to report missing, lost, stolen or counterfeit securities to the data base, (b) to confirm inquiry of the data base, and (c) to demonstrate compliance with Rule 17f–1. The Commission and the reporting institutions' examining authorities