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 April 25, 1998, through May 8, 1998. The last biweekly notice was published on May 6, 1998 (63 FR 25101).

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

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

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

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

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

By June 19, 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.

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

Date of amendment request: January 14, 1998.

Description of amendment request: The proposed amendments would change the Technical Specifications to allow replacement of the 125 volt direct current (DC) AT&T batteries with new Charter Power Systems, Inc. (C&D) batteries, and revise the crosstie loading limitation.

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

The replacement C&D battery has been selected to meet or exceed the design, functional, and operational requirements of those of the present AT&T battery, including crosstie load limitations. The C&D batteries are similar in design to the previously installed Gould batteries (e.g. electrolyte specific gravity and construction of the plates) except for capacity. The replacement C&D batteries have a significantly larger capacity than either the previously installed Gould, or the currently installed AT&T, batteries. This increased capacity can provide additional margin for future use. Also, the C&D batteries are qualified for a 20 year life and meet the latest applicable standards. The short circuit current provided by the C&D batteries is well within the interrupting capability of the existing DC system [c]ircuit breakers.

Additionally, the crosstie limit is increased to take advantage of the larger C&D battery capacity. The C&D batteries were sized based on having sufficient capacity to energize the design basis DC loads of an operating unit with the [Institute of Electrical and Electronics Engineers] IEEE–485 design margin while maintaining the desired limited DC load of 200 amps for a shutdown unit. This proposed change allows use of the C&D batteries' larger capacity.

Also, although adherence to the performance testing intervals stated in IEEE Std 450 could result in a planned shutdown and possible subsequent increase in the probability of occurrence of an accident (e.g. Turbine Trip), it would be part of a controlled and planned shutdown, therefore the increases would not be considered significant.

The overall design, function, and operation of the DC system and equipment has not been altered by these changes. The proposed changes do not affect any accident initiators of precursors and do not alter the design assumptions for the systems or components used to mitigate the consequences of an accident as analyzed in UFSAR [Updated Final Safety Analysis Report] Chapter 15. Therefore, there is no increase in the probability or consequences of an accident previously evaluated.

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

The replacement C&D batteries will provide the same function as those of the installed AT&T batteries and will be operated with the same types of operational controls. These limits include battery float terminal voltage, individual cell voltage and electrolyte specific gravity, and crosstie loading. Crosstie conditions are allowed under the present Technical Specifications. The crosstie limit is increased to take advantage of the larger C&D battery capacity. The remaining changes are administrative in nature or provide clarification to maintain consistency with other Technical Specifications.

The DC system and its equipment will continue to perform the same function and be operated in the same fashion. The proposed changes do not create any new or common failure modes. The proposed changes do not introduce any new accident initiators or precursors, or any new design assumptions for the systems or components used to mitigate the consequences of an accident. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated has not been created.

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

The replacement C&D batteries will meet or exceed the design, functional, and qualifications of the installed AT&T batteries. The proposed Technical Specification limitations for the C&D batteries are derived from the same methodology as the AT&T batteries with applied margins in accordance with IEEE 485. Increasing the crosstie loading limits takes advantage of the larger C&D battery capacity with its increased design margin. The proposed change to the crosstie loading limit will continue to conservatively envelope the postulated design requirements. The remaining changes are administrative in nature or provide clarification to maintain consistency with other Technical Specifications.

The inherent design conservatism of the DC system and its equipment has not been altered. The DC system and its equipment will continue to be operated with the same degree of conservatism. Therefore, there is no reduction in the margin of safety.

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

Local Public Document Room location: Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: May 27, 1997, as supplemented on August 1, 1997, and March 24, 1998.

Description of amendment request: The proposed amendments would revise Technical Specification Section 6, ''Administrative Controls,'' to incorporate revised organizational titles and would delete the Unit 1 License Condition 2.C.(30)(a) related to the function of the Shift Technical Advisor. In addition, the proposed amendments would change the submittal frequency of the Radiological Effluent Release Report from semiannually to annually. The proposed amendments will also make several administrative and editorial changes. The staff's proposed no significant hazards consideration determination for the requested change was published on July 30, 1997 (62 FR 40848).

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

The proposed changes do not affect any accident initiators or precursors and do not change or alter the design assumptions for systems or components used to mitigate the consequences of an accident. The proposed changes do not affect the design or operation of any system, structure, or component in the plant. There are no changes to parameters governing plant operation, and, no new or different type of equipment will be installed.

The proposed changes provide clarification, consistency with station procedures, programs, the Code of Federal Regulations (10 CFR), other Technical Specifications, and Improved Technical Specifications. These changes do not impact any accident previously evaluated in the UFSAR [Updated Final Safety Analysis Report]. There is no relaxation of applicable administrative controls. Those administrative requirements which have no effect on safe operation of the plant are eliminated.

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

The proposed changes do not affect the design or operation of any plant system, structure, or component. There are no changes to parameters governing plant operation, and, no new or different type of equipment will be installed.

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

The proposed changes do not affect the margin of safety for any Technical Specification. The initial conditions and methodologies used in the accident analyses remain unchanged; therefore, accident analyses results are not impacted. Plant safety parameters or setpoints are not affected. All responsibilities described in the Technical Specifications for administrative controls will continue to be performed by individuals possessing the requisite qualifications. Clarifications, relocations, and nomenclature changes neither result in a reduction of personnel responsibilities, nor do they cause a relaxation of programmatic controls. There are no resulting effects on plant safety parameters or setpoints.

Guidance has been provided in ''Final Procedures and Standards on No Significant Hazards Considerations,'' Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations. These proposed amendments most closely fit the example of a purely administrative change to the Technical Specifications to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

The proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings, or a significant relaxation of the bases for the limiting conditions for operations. The proposed change does not reduce the margin of safety as defined in the basis for any Technical Specification.

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

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

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: April 13, 1998.

Description of amendment request: Unreviewed Safety Question involving additional manual actions incorporated in new fire protection procedures as a result of a revised Appendix R Safe Shutdown Safety Analysis.

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) No significant increase in the probability or consequences of an accident previously evaluated is involved because of the following:

Two types of previously evaluated accidents are relevant to this criterion: (1) A fire; (2) other accidents evaluated in the Updated Final Safety Analysis Report. For these previously evaluated accidents, the change would not result in an increase in either their probabilities of occurrence or the consequences of their occurrence, for the following reasons:

The additional operator manual actions do not significantly change the probability or consequences of a fire. The likelihood of a fire is unchanged. Additional operations do not significantly change the fire loading nor introduce significant new ignition sources. The quantities and arrangement of combustible materials are not changed through additional manual actions.

The consequences of a fire are unchanged because operator manual actions serve to support the station's ability to achieve and maintain shutdown in the event of a fire.

Additional manual operations are for purposes of safe shutdown in the event of a fire in areas requiring alternate shutdown capability and do not impact other accident scenarios. Also, there is no increase in the predicted frequency of other accidents as a result of this change. Accordingly there is no significant change in the probability or consequences of other accidents previously evaluated because they are independent of this change in procedures for fire scenarios.

(2) The possibility of a new or different kind of accident from any accident previously evaluated is not created because:

The proposed change does not create the possibility of a new or different kind of accident from that previously evaluated for the Quad Cities Station. Although the number of manual actions increased and there may be some compression in the time for taking necessary actions relative to the current safe shutdown analysis and procedures, there is no significant change in the operation of plant equipment following the postulated fire event. The existing safe shutdown analysis already relies on operator manual actions which perform the same type of actions.

The overall approach and methodology to performing these operator actions are not significantly different from the prior approach and methodology. This proposed change does not involve an accident initiator or failure not previously considered. The results or effects of equipment malfunctions previously evaluated are unchanged as the result of potential operator errors. No new failures would occur, and no new modes of operation are introduced by the proposed changes.

Additional manual actions and the timing thereof provide a somewhat different demand on the plant equipment operators, but still provide an effective method for achieving and maintaining post-fire safe shutdown for areas requiring alternate shutdown capability. As such, the proposed changes do not create the possibility of a new or different kind of accident.

(3) No significant reduction in the margin of safety is involved because:

A change in the fire protection program does not result in a significant reduction in the margin of safety if the change does not result in a significant adverse impact on the plant's ability to achieve and maintain safe shutdown in the event of a fire. The proposed operator manual actions to achieve and maintain safe shutdown in a fire scenario do not significantly affect the capability or reliability of the equipment assumed to operate in the safety analysis.

The types of manual actions to be performed in support of Appendix R safe shutdown functions are not significantly different from those previously considered. The complexity of actions is not significantly changed. Indeed many of the additional actions are designed to provide additional protection from spurious operations which could result from a fire.

Any reduction in margin associated with changes in the time before which certain manual actions must occur is largely a result of re-analyses which incorporate conservatisms not previously considered. In total, the proposed changes do not adversely impact the capability to meet the requirements of Appendix R. Any reduction in margin associated with additional manual actions to achieve and maintain post fire safe shutdown in areas requiring alternate capabilities does not involve a significant reduction in margin.

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 are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

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

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Stuart A. Richards.

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, and April 20, 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 Commission 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. In DEC's March 9, March 20, and April 20, 1998, supplements, there are proposed changes that are beyond the scope of NUREG–1431, which were, thus, not covered by the staff's July 14, 1997, notice. 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. Table 3.3–3 of the current TS contains an entry regarding the Containment Pressure Control System, allowing an inoperable channel be placed in trip in 1 hour. DEC proposed to tighten this requirement such that the system supported by the inoperable channel be declared inoperable immediately. No changes to the design of the Containment Pressure Control System or other systems were proposed by DEC.

2. Table 4.3–1 of the Unit 1 current TS has a footnote (No. 13) that specifies a filter time constant of 1.5 seconds in the steam generator low-low level reactor trip circuitry. DEC proposed to delete this time constant since it was never used. No design changes to the instrumentation and control systems are involved.

3. Section 4.5.1.1.c of the current TS requires that power be removed from the accumulator isolation valve when the reactor coolant system pressure is greater than 2000 pounds per square inch gauge (psig). DEC proposed to make this requirement more restrictive, lowering this threshold to 1000 psig on the recommendation of the nuclear vendor, Westinghouse. No design changes to the accumulator system are involved.

4. Section 4.6.5.1.b.1 of the current TS requires that the boron concentration of the ice in the ice condenser be verified once every 9 months to be at least 1800 ppm. DEC proposed to relax the frequency from 9 months to 18 months on the basis that boron, in the form of sodium tetraborate, does not decrease in quantity even though the ice sublimates. No design changes to the ice condenser 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 were not factors in the 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 difference 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, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: April 20, 1998.

Description of amendment request: The Control Room Area Ventilation System (CRAVS) can be actuated by a number of ways, including by the engineered safety features actuation signal (ESFAS) when safety injection is also initiated. The only relationship between automatic actuation of the CRAVS and the ESFAS is through safety injection initiation, applicable in Modes 1, 2, 3, and 4. However, in Tables 3.3–3 and 4.3–2 of the units' Technical Specifications, regarding operability and surveillance requirements, the CRAVS automatic actuation has been erroneously specified for all modes (Modes 1, 2, 3, 4, 5, and 6). The licensee proposed to correct this error by the proposed amendment.

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

First Standard

Implementation of this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. The Control Room Area Ventilation System and ESFAS are not accident initiating systems; they are accident mitigating systems. Therefore, changing the mode requirements for the subject ESFAS functional unit cannot impact accident initiating probabilities. The technical justification associated with this proposed amendment shows that the current Technical Specification mode requirements for the subject functional unit are incorrect as written. The Control Room Area Ventilation System and ESFAS will remain fully capable of performing their design accident mitigation functions for the modes in which they are required. The Control Room Area Ventilation System operability requirement of Technical Specification 3/4.7.6 will continue to be met. 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 Control Room Area Ventilation System and ESFAS are not accident initiating systems. Correcting the mode requirements 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. Also, no changes are being made to the way in which the plant is operated, so no new failure mechanisms will be initiated.

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. Both the Control Room Area Ventilation System and the ESFAS will remain fully capable of performing their design functions for the modes in which they are required. Therefore, no safety margin will be significantly impacted.

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

Local Public Document Room location: 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), 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 letter dated March 9, 1998.

Description of amendment request: The three proposed changes are associated with DEC's application to convert to the Improved Technical Specifications. The first change would increase the surveillance interval for the boron concentration of the ice bed from once per 9 months, to every 18 months. This change is supported by operating experience data, establishes surveillance intervals that coincide with refueling outages, and minimizes containment entries during power operation. The second change would decrease the Reactor Coolant System pressure level at which power is removed from the accumulator isolation valve from 2000 pounds per square inch gauge (psig) to 1000 psig. This change is considered a more restrictive change, and is based on recommendations by Westinghouse Nuclear Safety Advisory Letter 97–003. The third change would revise the Turbine Trip and Feedwater Isolation function to include an initiation signal from the average-low temperature. This change is considered a more restrictive change, and is consistent with the plant design and safety analysis.

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 for each of the above proposed changes. The NRC staff has reviewed the licensee'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?

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 functions of these systems, and the systems were not factors in the 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?

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?

Margin of safety is associated with confidence in the design and operation of the plant. The proposed changes do not involve any change to the 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: J. Murrey Atkins Library, University of North Carolina at

Charlotte, 9201 University City Boulevard, North Carolina.

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

NRC Project Director: Herbert N. Berkow.

Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: April 9, 1998.

Description of amendment request: The proposed amendment would revise license condition 2.C(13) to allow Final Feedwater Temperature Reduction (FFWTR) at the River Bend Station, Unit No.1(RBS). FFWTR is to be used at the end of each fuel cycle to allow approximately fourteen additional effective full power days of operation.

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

1. The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated.

The abnormal operational occurrences or accidents analyzed in the SAR [Safety Analysis Report] have been examined for impact caused by partial feedwater heating during cycle extension or at coastdown condition. The limiting abnormal operation transients, including the Load Rejection with no Bypass (LRNBP) event and the Feedwater Controller Failure (FWCF) maximum demand event, Turbine Trip with No Bypass (TTNBP) and Pressure Regulator Failure Downscale (PRFD) have been analyzed based upon the core nuclear characteristic at end-of-cycle (EOC) conditions including the effects of increased core flow and the proposed reduction in feedwater temperature with an all-rods-out condition.

The LOCA (Loss of Coolant Accident), fuel loading error, rod drop accident, rod withdrawal error, overpressure protections and ATWS (anticipated transient without scram) analyses have been evaluated for the effects of reduced feedwater temperature operation and found acceptable. In addition, the case of the analyzed operational events the current fuel OLMCPR (operating limit minimum critical power ratio) and MAPLHGR (maximum average planar linear heat generation rate) limits bound those necessary for operation and therefore, are not affected by operation with FFWTR therefore, these events are bounded by the current RBS analysis. Because the accident results are acceptable and the current operating fuel limits are unaffected, the consequence of an event previously evaluated remains unaffected.

The probability of an accident is not affected by the proposed changes since no systems or equipment which could initiate an accident are affected. Therefore, the proposed changes do not significantly increase the probability or consequences of any previously evaluated accident.

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

The FFWTR mode of operation is functionally similar to operation with Feedwater Heaters Out of Service (USAR (Updated Safety Analysis Report) Section 15.1.7). All abnormal operational transients or accidents have been evaluated and the most limiting cases have been analyzed for applicability for the FFWTR operation. Limits on MAPLHGR and OLMCPR (including the power and flow dependent MCPR) which are included in the Core Operating Limits Report (COLR) as part of the normal reload licensing process will continue to assure that operations are within the assumptions, initial conditions and assumed power distribution and therefore will not create a new type of accident. The proposed changes do not involve new setpoints, new system interactions, or physical modifications to the plant. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previous analyzed.

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

The proposed changes do not involve any setpoint changes and would allow steady state power operation at off-rated feedwater temperature conditions as defined in current plant procedures. The transient and accidents described in the SAR are evaluated for effects caused by the reduced feedwater temperature of 100 (degrees) F. As described in Attachment 4 (to the April 9, 1998, amendment request), * * * the FWCF is the most limiting transient under such condition and the required OLMCPR for this event is bounded by the EOC OLMCPR limits set forth in the RBS COLR. The thermal limits MCPR and LHGR curves, and the MAPLHGR limits establish limits on power operation and thereby ensure that the core is operated within the assumptions and initial conditions of the transient or accident analyses.

Operation within these limits set forth by the MCPR limits, the LHGR limits and the MAPLHGR criteria will ensure that the margin of safety will be maintained to the same level described in the Technical Specifications Bases and the SAR. As a result the consequences of postulated transients or accidents are not increased.

The MCPR safety limit, mechanical performance limits and overpressure limits are not exceeded during any transient or postulated accident at normal feedwater temperature or at reduced feedwater temperature condition. Therefore, the proposed changes to allow partial feedwater heating for cycle extension do not involve a significant reduction in margin of safety.

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

Local Public Document Room location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803.

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

NRC Project Director: John N. Hannon.

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

Date of amendment request: April 27, 1998.

Description of amendment request: The proposed amendment would change the title of ''shift supervisor'' to ''shift manager'' in the Technical Specifications (TS).

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

(1) The proposed change replaces the title of ''shift supervisor'' with the title of ''shift manager'' as it pertains to the responsibilities of the position described in TS Section 5.1.2. The proposed change does not involve a change to the plant design or to the operation of the plant by qualified operators and senior operators. Although this change involves changes to the Operations department, individuals in those positions comprising the operating crews will continue to have to meet the same licensing, experience, training, and education requirements, notwithstanding the proposed change in the title of the individual with ultimate command authority in the main control room, from ''shift supervisor'' to ''shift manager.'' Therefore, the operation of CPS is not affected by this change. Further, as also noted, the proposed change does not affect plant design. It therefore would not affect systems, structures, or components important to safety, particularly those associated with the plant accident analyses. As a result, the proposed change does not affect any parameters or conditions that may contribute to the initiation of any accidents previously evaluated, nor does it affect the operation or response of systems, structures, or components assumed to mitigate postulated accidents that have been evaluated/analyzed. On this basis, IP has concluded that the proposed change will not result in a significant increase in the probability or consequences of any accident previously evaluated.

(2) As noted above, the proposed change does not involve a change to design or operation of the plant. As a result, the proposed change, which is only administrative in nature, cannot introduce

any new failure modes or precursors, parameters, or conditions that could cause or contribute to the initiation of any new accidents not previously evaluated. On this basis, IP has concluded that the proposed change will not create the possibility of a new or different kind of accident not previously evaluated.

(3) As noted above, the proposed change is an administrative change that involves no changes to plant design or operation, including the design or operation of systems, components, or structures important to safety. On this basis there are no margins of safety affected by the proposed change. As a result, IP has concluded that the proposed change will not result in a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Vespasian Warner Public Library, 120 West Johnson Street, Clinton, IL 61727.

Attorney for licensee: Leah Manning Stetzner, Vice President, General Counsel, and Corporate Secretary, 500 South 27th Street, Decatur, IL 62525.

NRC Project Director: Ronald R. Bellamy, Acting.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of amendment request: April 13, 1998.

Description of amendment request: The proposed amendment would amend the Technical Specifications to base the Limiting Condition for Operation for the fuel storage pool water level on a revised analysis of the fuel handling accident and on a new analysis for radiological shielding during movement of irradiated fuel.

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

The proposed change does not:

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

The proposed restrictions on the water level in the spent fuel pool has no impact on the probability or consequences of the remaining applicable design basis accidents. These restrictions are fulfilled by normal operating conditions, preserve initial conditions assumed in the analyses of postulated DBAs and ensure that the conditions of such DBAs are consistent with the analyses. Revised analysis was performed assuming a fuel handling accident occurs after the spent fuel fission products have decayed at least 1-year. The initial conditions assumed a minimum of 19 feet of water for iodine absorption. No credit was taken for control room or spent fuel pool ventilation filtration. The results of the revised analysis demonstrate that the projected doses resulting from a postulated fuel handling accident are insignificant in comparison to 10 CFR part 100 limits. Therefore, the proposed changes to the Technical Specifications do not involve any increase in the probability or consequences of any accident previously evaluated.

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

The proposed restrictions on the water level in the spent fuel pool are fulfilled by normal operating conditions and preserve initial conditions assumed in the analysis of postulated DBAs. These additional restrictions do not involve changes to any structure or equipment affecting the safe storage of irradiated fuel. The results of the revised analysis of a fuel handling accident demonstrate that the projected doses are insignificant in comparison to 10 CFR part 100 limits with a minimum of 19 feet of water for iodine absorption. In addition, maintaining this minimum water level will also provide sufficient shielding for personnel radiation protection during fuel movement. Therefore, the proposed changes to the Technical Specifications would not create the possibility of a new or different accident from any accident previously evaluated.

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

The proposed restrictions on the water level in the spent fuel pool preserve initial conditions assumed in the analyses of postulated DBAs and ensure that margins of safety contained in the analyses are maintained. The margin of safety for the fuel handling accident relates to the acceptance limit which the NRC approved during its review of the license. The fuel handling accident acceptance limit defined in the basis for the Maine Yankee Technical Specification (formerly specified as TS 3.13.D.10) is 10% of 10 CFR part 100 limits. A reduction in margin of safety occurs when the acceptance limit would no longer be met as a result of a proposed change. Since the acceptance limit is met, there is no reduction in margin of safety. The projected dose rates at the specified Fuel Storage Pool water level during fuel movement with a fuel assembly raised to its highest allowable height would result in personnel exposures within that previously assumed. There is no reduction in a margin of safety. The NRC acceptance limit which is that combination of occupancy time and dose rate that maintains personnel doses within 10 CFR 20.1201 limits is not exceeded. Therefore, the proposed changes to the MYTS would not involve a significant reduction in any margin of safety.

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

Local Public Document Room location: Wiscasset Public Library, High Street, PO Box 367, Wiscasset, ME 04578.

Attorney for licensee: Mary Ann Lynch, Esquire, Maine Yankee Atomic Power Company, PO Box 408, Wiscasset, ME 04578.

NRC Project Director: Seymour H. Weiss.

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

Date of amendment requests: March 2, 1998.

Description of amendment requests: The proposed amendments would remove the spent fuel pool special ventilation system operability-based restriction on crane operations in the spent fuel pool enclosure, while maintaining that restriction during spent fuel handling operations.

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

The proposed change does not affect any system that is a contributor to initiating events for previously evaluated anticipated operational occurrences and design basis accidents. Therefore, the proposed change will not increase the probability of any previously evaluated accident.

The proposed change does not impact the required availability of the spent fuel pool special ventilation system during spent fuel handling operations to mitigate the consequences of a fuel handling accident.

The proposed change does impact the required availability of the spent fuel pool special ventilation system during heavy load handling operations. However, this system is not required to mitigate the consequences of a heavy load dropping onto a spent fuel assembly. Such a requirement is not applicable at Prairie Island, because the heavy loads in the spent fuel pool enclosure are either handled with single-failure-proof cranes, rigging and plant procedures implementing Prairie Island commitments to NUREG–0612, or handled with spent fuel pool protective covers in place as described in the Prairie Island USAR (updated safety analysis report). The use of a single-failure-proof crane with rigging and procedures that implement the requirements of NUREG–0612 assures that the potential for a heavy load drop is extremely small and therefore consideration of the effects of heavy load drops is not required. Spent fuel pool covers prevent dropped loads* (*The covers do have a limit on the weight load they are analyzed to withstand.) from falling into the spent fuel pool and therefore consideration of the effects of heavy load drops is also not required. These actions taken to reduce the accident initiator probabilities to insignificant magnitudes negate any theoretically small increase in the consequence of a postulated heavy load drop accident resulting from the removal of a requirement to have one train of the spent fuel pool special ventilation system operable during crane operations. It is concluded in summary that the proposed change does not involve a significant increase in the consequences of any accident previously evaluated.

2. The proposed amendment[s] will not create the possibility of a new or different kind of accident from any accident previously analyzed.

The proposed change does impact the required availability of the spent fuel pool special ventilation system during heavy load handling operations. Load drop events over spent fuel are well understood and have been thoroughly evaluated. The proposed change will not create any new accident scenarios or create the possibility of a new or different kind of accident from any accident previously analyzed.

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

The proposed change does not impact the required availability of the spent fuel pool special ventilation system during spent fuel handling operations to mitigate the consequences of a fuel handling accident as described in the USAR. As a result the safety margin inherent in the 10 CFR part 100 dose limits is not reduced.

The proposed change does impact the required availability of the spent fuel pool special ventilation system during heavy load handling operations. However, this system is not required to mitigate the consequences of a heavy load dropping onto a spent fuel assembly because the potential for a load drop is extremely small. Provision of single-failure-proof equipment and compliance with the other requirements of NUREG–0612 (provide) a defense-in-depth approach to assure the safe handling of heavy loads which would otherwise be demonstrated to be safe by the deterministic analysis of the radiological effects of dropped heavy loads.

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

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

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

NRC Project Director: Cynthia A. Carpenter.

Pennsylvania Power and Light Company, Docket Nos. 50–387 and 50–388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: November 26, 1997.

Description of amendment request. The amendments to the Units 1 and 2 Technical Specifications Surveillance Requirement Section 4.7.1.3.a involve lowering the Ultimate Heat Sink (UHS) surveillance requirement maximum acceptable spray pond average temperature from 88 °F to 85 °F. This temperature is specified to assure that the post design basic accident (DBA) loss-of-coolant (LOCA) accident/loss of offsite power maximum UHS temperature will be maintained less than the UHS design temperature.

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.

This proposal does not involve an increase in the probability or consequences of an accident previously evaluated. The proposed change lowers the UHS temperature surveillance requirement so that the maximum post DBA UHS temperature is maintained less than that reported previously.

The UHS provides cooling to equipment and systems required for the safe shutdown of the plant following an accident with radiological consequence potential, such as a LOCA. The change in UHS initial temperature limit to 85 °F assures that the peak temperature will remain less than that reported previously. Therefore, the components cooled by the UHS will not be impacted and will be capable of performing their function as designed.

Based upon the analysis presented above, PP&L (Pennsylvania Power and Light Company) concludes that the proposed action does not involve an increase in the probability or consequences of an accident previously evaluated.

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

This proposal does not create the probability of a new or different type of accident from any accident previously evaluated. The proposed change lowers the UHS surveillance requirement temperature so that the maximum post DBA UHS temperature is maintained less than that reported previously. Therefore the operation of the components cooled by the UHS will not be impacted and will be capable of performing their design function.

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 the margin of safety.

The change does not involve a reduction in the margin of safety. The proposed change lowers the UHS surveillance temperature so that the maximum post DBA UHS temperature is maintained less than that reported previously. The margin of safety is unaffected since the maximum post DBA UHS temperature is not affected. Performance of equipment cooled by the UHS is unaffected.

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: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, PA 18701.

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

NRC Project Director: Robert A. Capra.

Pennsylvania Power and Light Company, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: March 16, 1998.

Description of amendment request: The proposed amendment would change the design basis and Technical Specifications to support the implementation of Hydrogen Water Chemistry.

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.

No Design Basis Event requiring functioning of the Main Steam Line Radiation monitors is defined in the FSAR. FSAR Section 7.2.1.1.4.2.(i) describing Main Steam Line Radiation monitoring states that for accidents resulting in gross fission product release ''the primary variables for trip initiation would be reactor vessel low level, reactor vessel high pressure, or high neutron flux''. Because the Main Steam Line Radiation Monitors [MSLRM] trip function is not used in any accident analyses this proposed setpoint change does not involve an increase in the probability or consequences of an accident previously evaluated.

In conformance to SRP 15.4.9, the analysis of the design basis CRDA assumed release of activity by leakage from an isolated condenser. As described in the FSAR, the main steam line radiation monitors will shut down the mechanical vacuum pump if operating and close its suction valves, thus isolating the condenser in the event of a Main Steam Line-High Radiation trip. Operation of the mechanical vacuum pump following burst failures of fuel rods insufficient to cause a main steam line radiation monitor trip was evaluated to better understand the potential impacts of raising the setpoint. Doses calculated under conservative conditions were small compared to the acceptance criteria for offsite dose of 25% of 10 CFR part 100 limits for offsite dose for the CRDA and 10 CFR 50 limits for control room dose.

Relocation of the Main Condenser Offgas Treatment System Explosive Gas Monitoring System requirements to the FSAR Section 16.3 (Technical Requirements Manual (TRM)) and procedures involves the use of an alternate regulatory process for controlling the instrumentation requirements. The change does not introduce any new modes of plant operation, make any physical changes, alter any operational setpoints, or change the surveillance requirements. Any change in the Main Condenser Offgas Treatment System Explosive Gas Monitoring System requirements would be evaluated pursuant to the requirements of 10 CFR 50.59.

The Technical Specifications, the Explosive Gas Mixture description contained in LCO/Surveillance 3.11.2.6/4.11.2.6 and associated bases will be moved and retained in TS Section 6.0 ''Administrative Controls''. The LCO specific limit and program details will be relocated to the FSAR Section 16.3 (TRM) and procedures and any changes controlled by the 10 CFR 50.59 process. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

These proposed changes to Technical Specifications do not require physical changes to instrument channels other than the Main Steam Radiation Monitor setpoint, or to any systems or component that interfaces with the instrumentation channels, therefore there is no change in the probability or consequences of any accident analyzed in the FSAR.

Finally, revising the TS index is an administrative change.

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

The proposed Main Steam Line Radiation setpoint change does not result in any design or physical configuration changes to the instrumentation channels. Operation incorporating the proposed change will not impair the instrumentation channels from performing as provided in the design basis.

Relocation of the Main Condenser Offgas Treatment System Explosive Gas Monitoring System requirements to the FSAR Section 16.3 (TRM) and procedures involves the use of an alternate regulatory process for controlling the instrumentation requirements. Therefore, the above change does not introduce any accident initiators as it does not involve any new modes of plant operation, make any physical changes, alter any operational setpoints, or change the surveillance requirements.

The Technical Specifications, the Explosive Gas Mixture description contained in LCO/Surveillance 3.11.2.6/4.11.2.6 and associated bases will be moved and retained in TS Section 6.0 ''Administrative Controls''. The LCO specific limit and program details involves the use of an alternate regulatory process for controlling the requirements.

Since the proposed changes to the Technical Specifications do not adversely impact the reliability of the safety required systems, no new or different kind of accident is created.

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

Raising the trip setpoint does not significantly reduce the sensitivity of the MSLRM's to alarm and initiate actions in response to gross fuel failures during power operation or to the design basis control rod drop accident. The source term assumed for the design basis CRDA greatly exceeds that required to initiate the main steam line high radiation trip. Raising the setpoint does not induce a delay in reaching the setpoint that would result in an increase in offsite dose from the design basis control rod drop accident. The delay time from fuel failure to monitor response is determined by the transport time for steam flow from the reactor vessel to the monitor location, which is not changed by either hydrogen water chemistry or by the monitor setpoint. Consequently, raising the trip setpoint will not result in an incremental increase in activity release, control room dose or offsite dose. Therefore, there is no reduction in the margin of safety for the design basis event.

The radiological consequences of small fuel rupture events, that would produce main steam line radiation levels below the proposed trip setpoint, are not significant. These postulated events were evaluated to better understand the potential impacts of raising the setpoint. The potential offsite doses from such an event, in the absence of a trip, would be small compared to the limits of 10 CFR part 50 for control room dose and to the acceptance criteria of 25% of 10 CFR part 100 limits for offsite dose from the design basis CRDA.

Relocation of the Main Condenser Offgas Treatment System Explosive Gas Monitoring System requirements to FSAR Section 16.3 (TRM) involves the use of an alternate regulatory process for controlling the instrumentation requirements. Any change in the Main Condenser Offgas Treatment System Explosive Gas Monitoring System requirements would be evaluated pursuant to the requirements of 10 CFR 50.59. Also, revising the TS index is an administrative change.

The Explosive Gas Mixture description contained in LCO/Surveillance 3.11.2.6/4.11.2.6 and associated bases will be moved and retained in TS Section 6.0 ''Administrative Controls''. The LCO specific limit and program details will be relocated to the FSAR Section 16.3 (TRM) and procedures and any changes controlled by the 10 CFR 50.59 process.

Therefore, the proposed 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: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, PA 18701.

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: February 16, 1998, as supplemented by letter dated April 2, 1998.

Description of amendment request: The proposed amendment request would revise Technical Specification 3/4.4.5, ''Steam Generators,'' and its Bases to allow the implementation of 1-volt voltage-based repair criteria for the steam generator tube support plate-to-tube intersections for Unit 2 in accordance with Generic Letter 95–05, and make related Unit 1 administrative changes for consistency of wording (the NRC had previously approved a similar 1-volt voltage-based repair criteria application for Unit 1). In addition, the proposed amendment would make an administrative change to Bases 3/4.4.6.2, ''Operational Leakage,'' to clarify that the allowable steam generator leakage specification applies to both Unit 1 and Unit 2.

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

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.

Structural Considerations

Industry testing of model boiler and operating plant tube specimens for free span tubing at room temperature conditions shows typical burst pressures in excess of 5000 psi for indications of ODSCC (outer diameter stress corrosion cracking) with voltage measurements at or below the current structural limit of 5.45 volts. One model boiler specimen with a voltage amplitude of 19 volts also exhibited a burst pressure greater than 5000 psi. Burst testing performed on one intersection pulled from STP (South Texas Project) Unit 1 in 1993 with a 0.51 volt indication yielded a measured burst pressure of 8900 psi at room temperature. Burst testing performed on another intersection pulled from STP Unit 1 in 1995 with a 0.48 volt indication yielded a measured burst pressure of 9950 psi at room temperature.

The next projected end-of-cycle (EOC) voltage compares favorably with the current structural limit considering the voltage growth rate for indications at STP. Using the methodology of Generic Letter 95–05, the structural limit is reduced by allowances for uncertainty and growth to develop a beginning-of-cycle (BOC) repair limit which should preclude EOC indications from growing in excess of the structural limit. The non-destructive examination (NDE) uncertainty to be applied per Generic Letter 95–05 is approximately 20%. The growth allowance will be 30%/EFPY [effective full power year] or a STP Unit 2-specific growth rate, to be calculated in accordance with Generic Letter 95–05, whichever is greater. Where the generator-specific growth rate exceeds both the Unit 2-specific average growth rate and 30%/EFPY, that generator-specific growth rate will be used for that generator. Each succeeding cycle upper voltage repair limit will also be conservatively established based on Generic Letter 95–05 methodology. By adding NDE uncertainty allowances and a growth allowance to the repair limit, the structural limit can be validated.

The upper voltage repair limit could be applied to bobbin coil voltages between the lower and upper repair limits to leave such indications in service independent of RPC [rotating pancake coil] confirmation. However, RPC-confirmed indications will be conservatively removed from service consistent with Generic Letter 95–05.

Leakage Considerations

As part of the implementation of voltage-based repair criteria, the distribution of EOC degradation indications at the TSP (tube support plate) intersections has been used to calculate the primary-to-secondary leakage which is bounded by the maximum leakage required to remain within the applicable dose limits of 10 CFR 100 (10 CFR part 100) and GDC (General Design Criterion) 19. This limit was calculated using the Technical Specification Reactor Coolant System (RCS) Iodine-131 transient spiking values consistent with NUREG–0800. Application of the voltage-based repair criteria requires the projection of postulated Main Steam Line Break (MSLB) leakage based on the projected EOC voltage distribution from the beginning of cycle voltage distribution. Projected EOC voltage distribution is developed using the most recent EOC eddy current results and a voltage measurement uncertainty. Draft NUREG–1477 and Generic Letter 95–05 require that all indications to which voltage-based repair criteria are applied must be included in the leakage projection.

The projected MSLB leakage rate calculation methodology prescribed in Generic Letter 95–05 will be used to calculate the EOC leakage. A Monte Carlo approach will be used to determine the EOC leakage, accounting for all of the bobbin coil eddy current test uncertainties, voltage growth, and an assumed probability of detection of 0.6. The fitted log-logistic probability of leakage correlation will be used to establish the MSLB leak rate for each cycle. This leak rate will be used for comparison with a bounding allowable leak rate in the faulted loop which would result in radiological consequences which are within the dose limits of 10 CFR part 100 for offsite doses and GDC 19 for control room doses. Due to the relatively low voltage levels of indications at STP to date and low voltage growth rates, it is expected that the actual calculated leakage values will be far less than this limit for each successive cycle.

Other Considerations

Those changes associated with grammatical corrections, deleting tube diameter information not applicable to South Texas, and applying the additional reporting requirements to Unit 2, are administrative and do not involve a change to, or the operation of, any safety-related system.

Therefore, implementation of voltage-based repair criteria does not adversely affect steam generator tube integrity and the radiological consequences will remain below the limits of 10 CFR part 100 and GDC 19. Operation of the facility in accordance with the proposed amendment would not result in any 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.

Implementation of the proposed steam generator tube voltage-based repair criteria for ODSCC at the TSP intersections does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism which could result in an accident outside of the region of the TSP elevations because the criteria do not apply outside the thickness of the TSPs. It is therefore expected that for all plant conditions, neither a single nor multiple tube rupture event would likely occur in a steam generator where voltage-based repair criteria has been applied.

Specifically, STP has implemented a maximum leakage rate of 150 gpd [gallons-per-day] per steam generator to help preclude the potential for excessive leakage during all plant conditions. The draft Reg Guide 1.121 criterion for establishing operational leakage rate limits governing plant shutdown is based upon leak-before-break (LBB) considerations to detect a free span crack before potential tube rupture as a result of faulted plant conditions. The 150 gpd limit is intended to provide for leakage detection and plant shutdown in the event of unexpected crack propagation outside the tube support plate thickness resulting in excessive leakage. Draft Reg Guide 1.121 acceptance criteria for establishing operating leakage limits are based on LBB considerations such that plant shutdown is initiated if permissible degradation is exceeded.

Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical degradation lengths. Additionally, the leak-before-break evaluation assumes that the entire crevice area is uncovered during the secondary side blowdown of a MSLB. Typically, it is expected for the vast majority of intersections, that only partial uncovery will occur. Therefore, the proximity of the TSP will enhance the burst capacity of the tube.

Steam generator tube integrity is continually maintained through inservice inspection and primary-to-secondary leakage monitoring. Any tubes falling outside the voltage-based repair criteria limits are removed from service.

Those changes associated with grammatical corrections, deleting tube diameter information not applicable to South Texas, and applying the additional reporting requirements to Unit 2, are administrative and do not involve a change to, or the operation of, any safety-related system.

Therefore, operating 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 use of the voltage-based bobbin probe for dispositioning ODSCC degraded tubes within TSP intersections is demonstrated to maintain steam generator tube integrity in accordance with the requirements of draft Reg Guide 1.121. Draft Reg Guide 1.121 describes a method acceptable to the NRC staff for meeting GDCs 14, 15, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable degradation are removed from service. Upon implementation of the criteria, even under the worst case conditions, the occurrence of ODSCC at the TSP elevation is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The EOC distribution of indications at the TSP elevations for each successive cycle will be confirmed to result in acceptable primary-to-secondary leakage during all plant conditions.

In addressing the combined effects of loss of coolant accident (LOCA) and safe shutdown earthquake (SSE) on the steam generators, as required by GDC 2, it has been determined that tube collapse may occur in the steam generators at some plants. This is not the case at STP Unit 2 as the TSPs do not become sufficiently deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to the combined effects of the leak-before-break-limited LOCA rarefaction wave and SSE loadings to affect tube integrity.

Because the leak-before-break methodology is applicable to the STP reactor coolant loop piping, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design of the plant. Implementation practices using the bobbin probe voltage based tube plugging criteria bounds Reg Guide 1.83, Rev. 1, considerations by:

(1) Using enhanced eddy current inspection guidelines consistent with those used by EPRI in developing the correlations. This provides consistency in voltage normalization.

(2) Performing a 100% bobbin coil inspection for all hot leg tube support plate intersections and all cold leg intersections down to the lowest cold leg tube support plate with known ODSCC indications at each cycle. The determination of the tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20% random sampling of tubes inspected over their full length, and

(3) Incorporating rotating pancake coil inspection for all tubes with bobbin voltages greater than 1.0 volt. This further establishes the principal degradation morphology as ODSCC.

Implementation of voltage-based repair criteria at TSP intersections will decrease the number of tubes which must be repaired at each subsequent inspection. Since the installation of tube plugs to remove ODSCC degraded tubes from service reduces the RCS flow margin, voltage-based repair criteria implementation will help preserve the margin of flow.

For each cycle the projected EOC primary-to-secondary leak rate allowed is bounded by a leak rate which limits the radiological consequences of a EOC MSLB to within the dose limits of 10CFR100 for offsite doses and 10CFR50, Appendix A, General Design Criteria (GDC) 19 for control room doses. Therefore, this change does not involve a significant reduction in the margin to safety.

The assessment of radiological consequences of an assumed steam line break applicable to STP Unit 1 was provided in Attachment 2 to ST–HL–AE–5359 on May 2, 1996. The submittal was made in response to questions from the Emergency Preparedness and Radiation Protection Branch and is applicable to Unit 2 as well. The staff concluded that the thyroid doses for the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and control room are within the acceptance criteria.

Those changes associated with grammatical corrections, deleting tube diameter information not applicable to South Texas, and applying the additional reporting requirements to Unit 2, are administrative and do not involve a change to, or the operation of, any safety-related system.

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

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

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

NRC Project Director: John N. Hannon.

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. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: April 24, 1998.

Brief description of amendment: The proposed amendment would revise Technical Specification 3/4.3.2, ''Engineered Safety Features Actuation System Instrumentation'' to allow a 2-hour surveillance interval to facilitate testing of the 6.9 kV Emergency Bus Undervoltage relays.

Date of publication of individual notice in the **Federal Register**: May 4, 1998 (63 FR 24574).

Expiration date of individual notice: May 18, 1998 for comments; June 3, 1998 for hearings.

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

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

Date of amendment request: April 7, 1998.

Description of amendment request: The proposed amendment would replace the pressurizer maximizer water inventory requirement with a pressurizer maximizer indicated level requirement.

Date of publication of individual notice in **Federal Register**: April 23, 1998 (63 FR 20219)

Expiration date of individual notice: May 26, 1998.

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

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

Date of amendment request: April 14, 1998.

Description of amendment request: The proposed amendment addresses an earlier identified condition relating to the plant operators' ability to meet the operator response time of 10 minutes assumed in Chapter 15 of the Final Safety Analysis Report for termination of an Inadvertent Safety Injection event.

Date of publication of individual notice in **Federal Register**: April 20, 1998 (63 FR 19532).

Expiration date of individual notice: May 20, 1998.

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

Pennsylvania Power and Light Company, Docket Nos. 50–387 and 50–388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: August 1, 1996, as supplemented on March 2, 1998.

Brief description of amendment request: The proposed amendments would revise the Technical Specifications as follows: (1.n.) Change the surveillance requirement frequency for verification that the average planar heat generation rate, minimum critical power ratio, linear heat generation rate, and average power range monitor gain and setpoint are within specified limits. Specifically, the frequency would be changed from within 12 hours after completion of a thermal power increase of at least 15 percent of rated thermal power (RTP) to once within 24 hours after greater than or equal to 25 percent RTP, 24 hours thereafter, and prior to exceeding 50 percent RTP; (2.o.) Change the surveillance requirement for the verification of the average power range monitor flow biased simulated thermal power-high time constant from 6 seconds plus or minus 1 second to less than 7 seconds. The lower limit of 5 seconds will be relocated to plant procedures since it is not a condition for operability of this reactor protection system function; (3.p.) Change the frequency of surveillance requirement for rod worth minimizer channel functional test; and (4.q.) Relocate the main steam line radiation monitor reactor protection system and isolation trips from the Technical Specifications to the plant-controlled Technical Requirements Manual.

Date of publication of individual notice in **Federal Register**: 1.n. April 27, 1998 (63 FR 20664); 2.o. April 27, 1998 (63 FR 20669); 3.p. April 27, 1998 (63 FR 20665); 4.q. April 27, 1998 (63 FR 20667).

Expiration date of individual notices: May 27, 1998 (all 4 notices).

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, PA 18701.

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

Date of application for amendment: February 27, 1998.

Brief Description of amendment: The amendment revised the Technical Specifications by revising the pressure-temperature and overpressure limits.

Date of publication of individual notice in **Federal Register**: March 9, 1998 (63 FR 11456).

Expiration date of individual notice: April 8, 1998.

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

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

Date of application for amendment: April 29, 1998.

Brief description of amendments: To amend the Watts Bar Nuclear Plant, Unit 1, Technical Specifications (TS) for the Hydrogen Mitigation System igniters. The amendment revises the TS limiting condition for operation, LCO 3.6.8, to provide temporary requirements for hydrogen ignitors to address the two Train A ignitors which are currently out of service.

Date of publication of individual notice in the **Federal Register**: May 7, 1998 (63 FR 25243).

Expiration date of individual notice: June 8, 1998.

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

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 Ch. I, which are set forth in the license amendment.

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

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

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

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

Date of application for amendments: December 4, 1996, as supplemented March 27, June 9, June 18, July 21, August 14, August 19, September 10, October 6, October 20, October 23, November 5, 1997, and January 12, January 28 and March 16, 1998.

Brief description of amendments: The amendments include the following:

1. The amendments added a new surveillance requirement (SR) 3.4.9.2 to the Improved Technical Specifications (ITS) which requires verification that the capacity of each required bank of pressurizer heaters is equal to or greater than 150 kW every 24 months.

2. The amendments changed the current TS applicability for the pressurizer safety valves for Mode 3 to specify that two safety valves shall be operable with all reactor coolant system (RCS) cold leg temperature ≤365 °F for Unit 1 and >301 °F for Unit 2. This is a less restrictive change.

3. As part of the conversion to the ITS, the amemdment changed a requirement that the power-operated relief valves be demonstrated operable by performing a channel functional test once per 31 days to once per 92 days.

4. The ITS LCO 3.4.1.3 eliminated the limit of 1 gpm total primary-to-secondary leakage through all steam generators and thus will only require a limit of 100 gallons per day through any one steam generator. This is an administrative change.

5. The amendment retains the requirement of SR 4.5.2.f.2 and specifies a frequency of 24 months. The amendment also adds a new SR 3.5.2.7 which requires verification that each LPSI pump stops on an actual or simulated actuation signal.

6. The amendment regarding the control room emergency ventilation system (CREVS) changes the surveillance interval from 18 months to 24 months (each refueling cycle) for SR 4.7.6.1.e.2 requires that each train of CREVS is demonstrated operable at least once every 18 months by verifying that on a control room high radiation test signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filter and charcoal adsorber banks and that both of the isolation valves in each duct and common exhaust duct, and isolation valve in the toilet exhaust area duct, close. The above change is less restrictive.

7. The amendment changes the surveillance interval regarding the control room emergency temperature system (CRETS) from 62 days on a staggered basis (one train every 31 days) to 24 months (each refueling interval) for SR 4.7.6.1.a.

8. The amendment changes the surveillance interval regarding the spent fuel pool exhaust ventilation system (SFPEVS) from 18 months to 24 months (each refueling interval) for SR 4.9.12.d. This is a less restrictive change.

9. The amendment changes the surveillance interval regarding the penetration room exhaust ventilation system (PREVS) from 18 months to 24 months (each refueling interval) for SR 4.6.6.1.d.2.

Date of issuance: May 4, 1998.

Effective date: As of the date of issuance to be implemented by August 31, 1998.

Amendment Nos.: 227 and 201.

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

Date of initial notice in **Federal Register**: March 6, 1998 (63 FR 11312) The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated May 4, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: October 29, 1997, as supplemented January 28 and April 20, 1998.

Brief Description of amendments: The amendments update the Technical Specification description of Control Rod Assemblies to allow for boron carbide or hafnium absorber materials, as approved by the NRC staff.

Date of issuance: April 27, 1998.

Effective date: April 27, 1998.

Amendment Nos.: 193 and 224.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments change the Technical Specifications.

Date of initial notice in **Federal Register**: December 17, 1997 (62 FR 66137) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 27, 1998.

No significant hazards consideration comments received: No.

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

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois

Date of application for amendments: November 7, 1997, as supplemented on March 24, 1998, and April 9, 1998.

Brief description of amendments: The amendments defer the next scheduled Type A containment integrated leak rate test for Byron, Unit 2, until the next refueling outage in 1999.

Date of issuance: May 8, 1998.

Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 102 and 102.

Facility Operating License Nos. NPF–37 and NPF–66: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 7, 1998 (63 FR 17036) The April 9, 1998, supplement provided clarifying information which did not change the staff's initial proposed no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 8, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010.

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

Date of application for amendments: September 26, 1997, as supplemented on April 7, 1998.

Brief description of amendments: The amendments revise Technical Specification 3.6.1.8 to prohibit the simultaneous opening of the drywell and suppression chamber purge system isolation valves and revise the surveillance requirements of TS 3/4.6.5.3, ''Standby Gas Treatment System'' to upgrade the filter testing methods to more current industry standards. This amendment approves only a portion of the request dated September 26, 1997. The remainder of the request will be addressed in separate correspondence.

Date of issuance: April 27, 1998.

Effective date: Immediately, to be implemented prior to startup of LaSalle, Unit 1, from the current outage and prior to restart of LaSalle, Unit 2, from the current outage.

Amendment Nos.: 125 and 110.

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

Date of initial notice in **Federal Register**: November 19, 1997 (62 FR 61840) The April 7, 1998, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 27, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: December 11, 1995, as supplemented January 18, September 3, October 2, October 18, and October 25, 1996, and March 28, 1997.

Brief description of amendment: The amendment revises administrative controls technical specifications (TS) and related surveillance requirements. Amendment 174, issued on October 31, 1996, provided a partial response to the licensee's request. This amendment completes action on the request.

NRC has also granted the request of Consumers Energy to withdraw a portion of its December 11, 1996, application. The proposed change would have deleted the requirements of current TS 4.5.4, ''Surveillance for Prestressing System,'' TS 4.5.5, ''End Anchorage Concrete Surveillance,'' and TS 4.5.8, ''Dome Delamination Surveillance,'' and replaced the requirements with proposed TS 6.5.5, ''Containment Structural Integrity Surveillance Program.'' However, by letter dated March 28, 1997, the licensee withdrew the proposed change. In addition, the staff has denied a portion of the amendment request regarding limitations on the dose rates resulting from radioactive material released in gaseous effluents to areas beyond the site boundary. A separate Notice of Partial Denial of Amendment to Facility Operating License and Opportunity for Hearing has been published in the **Federal Register**. For further details with respect to these actions, see the application for amendment dated December 11, 1996, as supplemented above, the licensee's letter dated March 28, 1997, 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: May 7, 1998.

Effective date: May 7, 1998, to be implemented within 60 days from date of issuance.

Amendment No.: 181.

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

Date of initial notice in **Federal Register**: September 20, 1996 (61 FR 49493) The October 2, October 18, and October 25, 1996, and March 28, 1997, 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 May 7, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: December 27, 1995, as supplemented September 4, October 18, and November 26, 1996, June 27 and November 21, 1997, and January 29, and April 10, 1998.

Brief description of amendment: The amendment revises specification requirements and associated bases regarding the electrical power systems to closely emulate the Standard Technical Specifications for Combustion Engineer Plants, NUREG–1432, Revision 1.

Date of issuance: April 29, 1998.

Effective date: The license amendment is effective as of the date of issuance with full implementation within 60 days after Cold Shutdown following completion of the 1998 refueling outage, but no later than October 2, 1998.

Amendment No.: 180.

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

Date of initial notice in **Federal Register**: April 9, 1997 (62 FR 17229) The June 27 and November 21, 1997, and January 29 and April 10, 1998, letters provided clarifying information that was 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 April 29, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: September 18, 1997, as supplemented by letter dated February 24, 1998.

Brief description of amendment: The amendment decreases the safety limit for the minimum critical power ratio (MCPR) from 1.12 to 1.11 for two recirculation loop operation and from 1.14 to 1.12 for single recirculation loop operation in Technical Specification (TS) 2.1.1.2. Because the proposed amendment is for Cycle 10 operation, the amendment would also revise the footnotes to TSs 2.1.1.2 and 5.6.5 to state that the MCPR values and the items 19 and 20, two topical reports being added to the core operating limits report in TS 5.6.5, are ''applicable only for Cycle 10 operation.'' Cycle 10 operation begins at the plant restart from the current refueling outage No. 9.

Date of issuance: May 8, 1998.

Effective date: May 8, 1998.

Amendment No: 136.

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

Date of initial notice in **Federal Register**: October 22, 1998 (62 FR 54872) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 8, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 4, 1997.

Brief description of amendment: To revise the Final Safety Analysis Report (FSAR) and the Improved Technical Specification (TS) Bases to reflect the modified reactor building fan control logic for fan AHF–1C.

Date of issuance: April 29, 1998.

Effective date: April 29, 1998.

Amendment No.: 166.

Facility Operating License No. DPR–72: Amendment revised the updated FSAR and TS Bases.

Date of initial notice in **Federal Register**: November 13, 1997 (62 FR 60921) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 29, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 32629.

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

Date of application for amendment: July 22, 1997.

Brief description of amendment: The amendment will incorporate a recent evaluation of a postulated inadvertent opening of a main steam safety valve into the current licensing basis for St. Lucie Unit 1.

Date of Issuance: April 30, 1998.

Effective Date: April 30, 1998.

Amendment No.: 154.

Facility Operating License No. NPF–16: Amendment revised the Updated Final Safety Evaluation Report.

Date of initial notice in **Federal Register**: August 27, 1997 (62 FR 45457) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 30, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 6, 1998, as supplemented March 30, March 31, and April 13, 1998.

Brief description of amendments: The amendments update the Technical Specification heatup and cooldown rate curves and extend their reactor vessel fluence limit from the current 20 effective full power years (EFPYs) to a new value of 35 EFPYs, incorporate into Technical Specifications the use of a Pressure and Temperature Limits Report, and change the power-operated relief valves temperature requirement for operability.

Date of issuance: May 4, 1998.

Effective date: May 4, 1998, with full implementation within 30 days.

Amendment Nos.: 135, 127.

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

Date of initial notice in **Federal Register**: March 27, 1998 (63 FR 14972) The March 30, March 31, and April 13, 1998, letters provided clarifying information and updated Technical Specification pages 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 amendments is contained in a Safety Evaluation dated May 4, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: February 9, 1998, as supplemented April 8 and 24, 1998.

Brief description of amendment: The amendment revises the minimal critical power ratio safety limits for operation Cycle 8.

Date of issuance: May 4, 1998.

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

Amendment No.: 127.

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

Date of initial notice in **Federal Register**: February 25, 1998 (63 FR 9613) The April 8 and 24, 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 May 4, 1998.

No significant hazards consideration comments received: No.

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

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

Date of amendments request: February 14, 1997, as supplemented by letters dated June 20, August 5, September 22, November 19, December 9, December 17, and December 31, 1997, January 23, February 12, February 26, March 3, March 6, March 16, April 3, April 13, and two letters on April 17, 1998.

Brief Description of amendments: The amendments change the maximum reactor core power level for facility operation from 2652 megawatts-thermal (MWt) to 2775 MWt for the Joseph M. Farley Nuclear Plant, Units 1 and 2. The amendments also approve changes to the Technical Specifications to implement uprated power operation.

Date of issuance: April 29, 1998.

Effective date: As of the date of issuance to be implemented prior to entering Mode 4 for Cycle 16 (fall 1998) for Unit 1 and prior to entering Mode 4 for Cycle 13 (spring 1998) for Unit 2.

Amendment Nos.: Unit 1—137; Unit 2—129.

Facility Operating License Nos. NPF–2 and NPF–8: Amendments revise the Technical Specifications, Operating Licenses, and adds a new Appendix C to the Operating Licenses.

Date of initial notice in **Federal Register**: October 8, 1997 (62 FR 52588) The November 19, December 9, December 17, and December 31, 1997, January 23, February 12, February 26, March 3, March 6, March 16, April 3, April 13, and two letters on April 17, 1998, provided additional and clarifying information that did not change the scope of the February 14, 1997, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 29, 1998, and an Environmental Statement was prepared and dated April 17, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Berdache Street, Post Office Box 1369, Dothan, Alabama.

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

Date of amendment request: April 9, 1998 (TXX–98107).

Brief description of amendments: The proposed amendment would allow on a one time basis, the verification of the proper operation of the Unit 2 load shed seal-in contacts and the diesel generator trip bypass contacts at power and crediting performance of Surveillance Requirements (SR) 4.8.1.1.2f.4) and 4.8.1.1.2f.6), at power as opposed to ''during shutdown'' as currently required by those SR. The proposed amendment would also allow on a one time basis the verification of the proper operation of the Unit 2 lockout relays and contacts to be deferred until the startup from the Unit 2 fourth refueling outage (2RFO4) or earlier outage to at least MODE 3.

Date of issuance: May 8, 1998.

Effective date: May 8, 1998.

Amendment Nos.: Unit 1—Amendment No. 59; Unit 2—Amendment No. 45.

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

Date of initial notice in **Federal Register**: April 20, 1998, (63 FR 19534). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 8, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 8, 1997, as supplemented by letter dated November 10, 1997.

Brief description of amendment: The amendment revises the feedwater isolation engineered safety feature actuation system (ESFAS) functions in Technical Specification Tables 3.3–3, 3.3–4, and 4.3–2.

Date of issuance: April 23, 1998.

Effective date: April 23, 1998, to be implemented within 30 days from the date of issuance.

Amendment No.: 126.

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

Date of initial notice in **Federal Register**: December 17, 1997 (62 FR 66144) The November 10, 1997, supplemental letter provided additional clarifying information that did not change the staff's original no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 23, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Missouri—Columbia, Elmer Ellis Library, Columbia, Missouri 65201–5149.

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

Date of amendment request: February 4, 1998.

Brief description of amendment: The amendment would revise Technical Specification 3.2.4, ''Quadrant Power Tilt Ratio,'' (QPTR) and its associated Bases to reflect (1) a change in the action for determining QPTR when QPTR is above 1.02, (2) a change in the completion time for resetting the power range neutron flux-high trip setpoints after QPTR is determined to be above 1.02, and (3) deletion of actions requiring QPTR to be restored within 24 hours, QPTR to be verified during a return to power operation, resetting the power range neutron flux-high trip setpoint to less than 55 percent following a power reduction to 50 percent reactor thermal power or below, and actions for QPTR in excess of 1.09.

Date of issuance: April 27, 1998.

Effective date: April 27, 1998, to be implemented within 60 days from the date of issuance.

Amendment No.: 116.

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

Date of initial notice in **Federal Register**: March 25, 1998 (63 FR 14489) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 27, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621.

Dated at Rockville, MD., this 13th day of May 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 98–13223 Filed 5–19–98; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF MANAGEMENT AND BUDGET

Cumulative Report on Rescissions and Deferrals

May 1, 1998.

This report is submitted in fulfillment of the requirement of Section 1014(e) of the Congressional Budget and Impoundment Control Act of 1974 (Public Law 93–344). Section 1014(e) requires a monthly report listing all budget authority for the current fiscal year for which, as of the first day of the month, a special message had been transmitted to Congress.

This report gives the status, as of May 1, 1998, of 24 rescission proposals and eight deferrals contained in two special messages for FY 1998. These messages were transmitted to Congress on February 3 and February 20, 1998.

Rescissions (Attachments A and C)

As of May 1, 1998, 24 rescission proposals totaling $20 million had been transmitted to the Congress. Congress approved 21 of the Administration's rescission proposals in P.L. 105–174. A total of $17.3 million of the rescissions proposed by the President was rescinded by that measure. Attachment C shows the status of the FY 1998 rescission proposals.

Deferrals (Attachments B and D)

As of May 1, 1998, $3,293 million in budget authority was being deferred from obligation. Attachment D shows the status of each deferral reported during FY 1998.

Information From Special Messages

The special messages containing information on the rescission proposals and deferrals that are covered by this cumulative report are printed in the editions of the **Federal Register** cited below:

63 FR 7004, Wednesday, February 11, 1998

63 FR 10076, Friday, February 27, 1998

Franklin D. Raines,

Director.

Attachments

ATTACHMENT A.—STATUS OF FY 1998 RESCISSIONS

[In millions of dollars]

	Budgetary resources
Rescissions proposed by the President	20.1
Rejected by the Congress	
Amounts rescinded by P.L. 105–174, the FY 1998 Supplemental Appropriations and Rescissions Act	−17.3
Currently before the Congress	2.8

ATTACHMENT B.—STATUS OF FY 1998 DEFERRALS

[In millions of dollars]

	Budgetary resources
Deferrals proposed by the President	4,833.0
Routine Executive releases through May 1, 1998 (OMB/Agency releases of $1,540.1 million, partially offset by cumulative positive adjustment of $0.3 million)	−1,539.8
Overturned by the Congress	....
Currently before the Congress	3,293.2