implementation of the merger agreement between UEC and CIPSCO, which provides for UEC to become a wholly-owned subsidiary of the newly formed Ameren Corporation, does not represent a ''significant change.''

Dated at Rockville, Maryland, this 16th day of October 1997.

For the Nuclear Regulatory Commission.

Samuel J. Collins,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 97–28000 Filed 10–21–97; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–483]

In the Matter of Union Electric Company (Callaway Plant, Unit 1); Order Approving Application Regarding the Corporate Merger Agreement Between Union Electric Company and Cipsco Incorporated To Form a Holding Company

I

Union Electric Company (UEC) is sole owner of Callaway Plant, Unit 1. UEC holds Facility Operating License No. NPF–30 issued by the U.S. Nuclear Regulatory Commission (NRC) pursuant to Part 50 of Title 10 of the Code of Federal Regulations on October 18, 1984. Under this license, UEC has the authority to own and operate Callaway Plant, Unit 1. Callaway Plant is located in Callaway County, Missouri.

II

By letter dated February 23, 1996, as supplemented by letters dated April 24, 1996, and November 15, 1996, UEC informed the Commission that it had entered into a merger agreement with CIPSCO Incorporated (CIPSCO) which would provide for UEC to become a wholly-owned operating company of Ameren Corporation (Ameren). Ameren was formed to implement the merger agreement, and is presently owned equally by UEC and CIPSCO. Under the merger agreement, current holders of UEC common stock and holders of CIPSCO common stock will become holders of common stock in Ameren. UEC requested, to the extent necessary, the Commission's approval, pursuant to 10 CFR 50.80. Notice of this application for approval was published in the **Federal Register** on June 10, 1996 (61 FR 29434), and an Environmental Assessment and Finding of No Significant Impact was published in the **Federal Register** on November 22, 1996 (61 FR 59469).

Under 10 CFR 50.80, no license shall be transferred, directly or indirectly, through transfer of control of the license, unless the Commission shall give its consent in writing. Upon review of the information submitted in the letter of February 23, 1996, as supplemented by letters dated April 24, 1996, and November 15, 1996, and other information before the Commission, the NRC staff has determined that consummation of the merger agreement between UEC and CIPSCO, resulting in UEC becoming a wholly-owned subsidiary of a holding company, Ameren, will not affect the qualifications of UEC as holder of the license for Callaway Plant, and that the transfer of control of the license, to the extent effected by the consummation of the merger agreement between UEC and CIPSCO, is otherwise consistent with applicable provisions of law, regulations, and orders issued by the Commission, subject to the conditions set forth herein. These findings are supported by the Safety Evaluation dated October 16, 1997.

III

Accordingly, pursuant to Section 161b, 161i, 161o, and 184 of the Atomic Energy Act of 1954, as amended, 42 USC 2201(b), 2201(i), 2201(o) and 2234, and 10 CFR 50.80, *It Is Hereby Ordered* that the Commission approves the application regarding the merger agreement between UEC and CIPSCO, under which Ameren will become the holding company of UEC, subject to the following: (1) UEC shall provide the Director of the Office of Nuclear Reactor Regulation a copy of any application, at the time it is filed, to transfer (excluding grants of security interests or liens) from UEC to its proposed parent or to any other affiliated company, facilities or other assets for the production, transmission, or distribution of electric energy having a depreciated book value exceeding ten percent (10%) of UEC's consolidated net utility plant, as recorded on UEC's books of account; and (2) should the merger agreement between UEC and CIPSCO not be implemented by September 30, 1998, this Order shall become null and void, provided, however, on application and for good cause shown, such date may be extended.

This Order is effective upon issuance.

IV

By November 21, 1997, any person adversely affected by this Order may file a request for a hearing with respect to issuance of the Order. Any person requesting a hearing shall set forth with particularity how that interest is adversely affected by this Order and shall address the criteria set forth in 10 CFR 2.714(d).

If a hearing is to be held, the Commission will issue an order designating the time and place of such hearing.

The issue to be considered at any such hearing shall be whether this Order should be sustained.

Any request for a hearing 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 11555 Rockville Pike, Rockville, Maryland between 7:45 am and 4:15 pm Federal workdays, by the above date. Copies should be also sent to the Office of the General Counsel, and to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to Gerald Charnoff, Esquire/Thomas A. Baxter, Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N. Street, N.W., Washington, D.C. 20037, attorneys for UEC.

For further details with respect to this Order, see the application dated February 23, 1996, and supplemental letters dated April 24, 1996 and November 15, 1996, which 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 located at the Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

Dated at Rockville, Maryland, this 16th day of October 1997.

For the Nuclear Regulatory Commission.

Samuel J. Collins,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 97–28001 Filed 10–21–97; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from September 29, 1997, through October 9, 1997. The last biweekly notice was published on October 8, 1997 (62 FR 52578).

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

By November 21, 1997, 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 50-454 and STN 50-455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois

Date of amendment request: April 7, 1997, as supplemented on August 7, 1997.

Description of amendment request: The proposed amendment would revise the plants' technical specifications to permit replacement of the 125 volt dc Gould batteries with new C&D Charter Power Systems, Inc., batteries.

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 or 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 Gould battery, including crosstie load limitations. The C&D batteries are similar in design to the 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 the Gould batteries, which 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 circuit 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 for an operating unit with the 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. 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 or 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 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 functions as those of the installed Gould 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 functions and be operated in the same fashion. The proposed change does 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 qualification requirements [of] those of the installed Gould batteries. The proposed Technical Specification limitations for the C&D batteries are derived from the same methodology as the Gould batteries with applied margins in accordance with IEEE-485. Increasing the crosstie loading limit 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: Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010

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

NRC Project Director: Robert A. Capra

Commonwealth Edison Company, Docket Nos. STN 50-454 and STN 50-455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois Docket Nos. STN 50-456 and STN 50-457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of amendment request: June 30, 1997, as supplemented on September 25, 1997.

Description of amendment request: The proposed amendment would revise the plants' technical specifications to permit the licensee to take credit for soluble boron in spent fuel storage pool water to maintain an acceptable margin of subcriticality.

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

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

The following accidents have been specifically evaluated relative to the SFP [spent fuel pool]: fuel assembly drop, accidental misloading of spent fuel assemblies into the SFP racks, and loss of normal cooling.

There is no increase in the probability of a fuel assembly drop accident in the SFP when considering the presence of soluble boron in the SFP water for criticality control. The handling of the fuel assemblies in the SFP has previously been performed in borated water. The criticality analysis shows the consequences of a fuel assembly drop accident in the SFP are not affected when considering the presence of soluble boron.

There is no increase in the probability of the accidental misloading of spent fuel assemblies into the SFP racks when considering the presence of soluble boron in the pool water for criticality control. Fuel assembly placement will continue to be controlled in accordance with approved fuel handling procedures and the spent fuel storage configuration limitations. Periodic surveillances of the SFP inventory (physical inventory and piece counts) are performed in accordance with station procedures. These surveillances ensure physical SFP inventory verification is performed at least once per year and in a timely manner upon completion of fuel movement in the SFP. The addition of credit for decay time in the spent fuel pool in determining allowable storage requirements is an extension of the reactivity equivalencing methodologies used for burnup credit in WCAP-14416-NP-A, ''Westinghouse Spent Fuel Rack Criticality Analysis Methodology,'' Revision 1, November 1996.

There is no increase in the consequences of the accidental misloading of spent fuel assemblies into the SFP racks because criticality analyses demonstrate that the pool will remain subcritical following an accidental misloading if the pool contains an adequate boron concentration. The proposed TS limitations and surveillance frequency will ensure that an adequate SFP boron concentration is maintained.

There is no increase in the probability of the loss of normal cooling to the SFP water when considering the presence of soluble boron in the pool water for subcriticality control since a high concentration of soluble boron has previously been maintained in the SFP water. A loss of normal cooling to the SFP water causes an increase in the temperature of the water passing through the stored fuel assemblies. This causes a decrease in water density which would result in a decrease in reactivity when Boraflex neutron absorber panels are present in the racks. However, since the proposed change does not consider Boraflex to be present in the racks, and the SFP water has a high concentration of boron, a density decrease causes a positive reactivity addition. [The] consequences of this accident are bounded by the misloaded assembly analysis. Because adequate soluble boron will be maintained in the SFP water, the consequences of a loss of normal cooling to the SFP will not be increased.

The proposed 48 hour surveillance frequency will be used to verify the boron concentration is within the initial assumptions of the criticality analysis. The current frequency of 24 hours was based on the sampling frequency for reactor coolant system (RCS) shutdown margin in Mode 5. A dilution of the SFP to a k_{eff} greater than 0.95 would take a much longer time than an RCS dilution resulting in loss of shutdown margin. This is due to the larger SFP volume compared to the RCS volume, and the turnover rate of water in the SFP is much less due to the lack of large dilution sources for the SFP. The 48 hour sampling frequency is sufficient based on operating experience, and based on the fact that significant changes in the boron concentration in the spent SFP are difficult to produce without detection, due to the large inventory of water. Soluble boron concentration reduction requires the inflow and outflow of large volumes of water which are readily detected by SFP and fuel handling building sump high level alarms, flooding in the fuel handling building or by normal operator rounds through the SFP area (once every eight hours), allowing adequate time for operator intervention prior to exceeding a k_{eff} of 0.95. Therefore, consequences of an accident previously evaluated are not increased by the change in surveillance frequency.

The format revisions to Specification 5.6.1.1 and reference to the report containing the specific NRC-approved criticality methodology in Specification 6.9.1.10 are administrative in nature and will not result in an increase in the probability or consequences of an accident previously evaluated.

Therefore, based on the above analysis, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The results of criticality accident analyses in the SFP are discussed in the UFSAR [Updated Final Safety Analysis Report] and in Criticality Analysis Reports associated with previous licensing activities. Specific accidents considered include fuel assembly drop, accidental misloading of spent fuel assemblies into the SFP racks, and loss of normal cooling.

LCO 3.9.1, ''BORON CONCENTRATION,'' contains limitations on the boron concentration in the filled portions of the reactor coolant system and the refueling canal during Mode 6. ComEd has maintained soluble boron in the SFP at all times and has imposed administrative limits on the SFP boron concentration, due in part to this requirement. LCO 3.9.11 establishes specific boron concentration requirements for the SFP water consistent with the results of the new criticality analysis based on the NRC-approved methodology of WCAP-14416-NP-A, ''Westinghouse Spent Fuel Rack Criticality Analysis Methodology,'' Revision 1, November 1996. Credit is also taken for radioactive decay time of the spent fuel.

Since soluble boron has always been maintained in the SFP water and is currently controlled administratively, the implementation of this requirement will have little effect on normal pool operations and maintenance. The implementation of the proposed limitations on the SFP boron concentration will only result in a requirement to verify boron concentration of the SFP water every 48 hours rather than every 24 hours. Sampling every 48 hours is sufficient to verify the SFP boron concentration meets the assumptions of the criticality analysis.

Because soluble boron has always been present in the SFP and has been administratively controlled, a dilution of the SFP soluble boron has always been a possibility. As shown in the SFP dilution evaluation performed for Byron and Braidwood, a dilution of the SFP which could increase the rack k_{eff} to greater than 0.95 (i.e., which could reduce the required margin to criticality) is not a credible event.

Therefore, the implementation of the proposed limitations on the SFP boron concentration and surveillance frequency will not result in the possibility of a new kind of accident.

The proposed change to Specification 5.6.1.1 identifies the requirements for the spent fuel rack storage configurations. The proposed changes relate to the criteria for determining the storage configuration. Since the proposed SFP storage configuration limitations will be similar to those currently in the Byron and Braidwood TS, these limitations will not have any significant effect on normal SFP operations and maintenance and will not create any possibility of a new or different kind of accident. Verifications will continue to be performed to ensure that the SFP loading configuration meets specified requirements.

The format revisions to Specification 5.6.1.1 and reference to the report containing the specific NRC-approved criticality methodology in Specification 6.9.1.10 are administrative in nature and will not create the possibility of a new [or] different kind of accident.

As discussed above, there is no significant change in plant configuration or equipment and the proposed changes will not create the possibility of a new or different kind of accident.

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

The proposed TS changes and the resulting spent fuel storage operating limits will provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. These limits are based on a plant specific criticality analysis performed in accordance with the NRC-approved Westinghouse spent fuel rack criticality analysis methodology (WCAP-14416-NP-A). Credit is also taken for radioactive decay time of the spent fuel.

Soluble boron credit provides significant negative reactivity in the SFP such that the k_{eff} is maintained less than or equal to 0.95. The proposed surveillance frequency will be used to verify the boron concentration is within the initial assumptions of the criticality analysis. A storage configuration has also been defined, with a 95-percent probability at a 95-percent confidence level, that ensures the spent fuel rack k_{eff} will be less than 1.0 with no credit for soluble boron or Boraflex panels in the racks. In addition to soluble boron credit, credit is taken for fuel assembly burnup, decay time, and IFBAs [Integral Fuel Burnable Absorber] when determining assembly storage requirements.

The loss of substantial amounts of soluble boron from the SFP which could lead to exceeding a k_{eff} of 0.95 has been evaluated and shown not to be credible. These evaluations show that the dilution of the SFP boron concentration from 2000 ppm to 550 ppm is not credible and that the spent fuel rack k_{eff} will remain less than 1.0 when flooded with unborated water.

The format revisions to Specification 5.6.1.1 and reference to the report containing the specific NRC-approved criticality methodology in Specification 6.9.1.10 are administrative in nature and will not result in a significant reduction in the plant's margin of safety.

Therefore, the proposed changes in this license amendment will not result in a significant reduction in the plant's margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the 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: Robert A. Capra

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

Date of amendment request: August 12, 1997

Description of amendment request: The proposed amendments would remove a Technical Specification surveillance requirement to verify that sediment deposition within the lake screenhouse is not greater than one foot in thickness. Control of sediment accumulation in the lake screenhouse would be accomplished through the Service Water Performance Monitoring Program.

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 identified because:

Surveillance's [sic] to fully verify [that] the Ultimate Heat Sink contains enough water to perform its design function will continue. All cleanliness issues associated with ensuring operability of Core Standby Cooling System - Equipment Cooling Water System (CSCS-ECWS) equipment will be performed under the Service Water Performance Monitoring Program, which meets GL 89-13 [≥Service Water System Problems Affecting Safety-Related Equipment≥] recommended actions. By performing these inspections per GL 89-13, LaSalle will ensure that there is no build up of sediment, which could hinder or impede the design operation of any safety or non-safety related equipment which takes a suction from the service water tunnel. Based on the nature of sediment, where it collects, and system design, the CSCS-ECWS will be available if called upon or started to respond in case of an accident for equipment cooling and long term cooling.

At no time, during approximately fourteen years of LaSalle operation, has sediment built up or accumulated either in front of the inlet to the CSCS cooling water screen bypass supply line or the six 36-inch normal tunnel supply lines in such a manner that the flow of water through these lines could have been reduced or blocked. Instead, loose sediment collects in quiescent areas near the traveling screens, the north end of the Service Water Tunnel, under the outlets of the 36-inch normal tunnel supply lines in the service water tunnel, and downstream of the butterfly isolation valve in the 54 inch CSCS cooling water screen bypass supply line. The sediment that collects in the service water tunnel does not build up in a manner such that CSCS-ECWS, non-essential station service water, or fire pump suctions from the tunnel are affected, based on inspections since 1992.

The CSCS equipment cooling bypass valve, OE12-F300, is the manual butterfly valve in the CSCS cooling water screen bypass supply line. The bypass valve is being added to the ASME Section XI Inservice Testing Program to cycle the valve quarterly. This valve cycling will help maintain sediment level in the bypass line at a low level due to flow through the line while the valve is not fully closed and thus assure the bypass line remains available. The flow is created due to the differential pressure across the circulating water traveling screens with circulating water pumps in operation.

Therefore, neither essential nor non-essential service water will be lost due to sediment. Neither the probability nor the consequences of an accident are increased by the deletion of SR 4.7.1.3.c.

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

Inspections for sedimentation will continue to be required by LaSalle's Service Water System Performance Monitoring Program per GL 89-13, to ensure continued operability of Core Standby Cooling System-Equipment Cooling Water System (CSCS-ECWS). The Ultimate Heat Sink operability requires assurance of a specific volume of water to provide cooling for at least 30 days for long term cooling following an accident. The public will be protected by the safety analysis in place by the fact that the safety and non-safety related equipment which take a suction from the service water tunnel will not be impaired by sediment. Therefore, there will be no possibility of a new or different kind of accident from any accident previously evaluated.

3) Involve a significant reduction in the margin of safety because:

The Ultimate Heat Sink continues to be demonstrated Operable by verifying a sufficient volume of water per TS SR 4.7.1.3.a and 4.7.1.3.b. Equipment operability will still be required per Technical Specifications 3/4.7.1.1 and 3/4.7.1.2 for the CSCS-ECWS systems. Sedimentation in the lake screenhouse is a maintenance/cleanliness issue addressed by the LaSalle Service Water Performance Monitoring Program. The program ensures equipment operability by both inspection for and removal of sedimentation and chemical control with a biocide to limit the growth of biological material and silt dispersant to help keep silt in the flow stream from coagulating. Therefore, there is minimal or no reduction in the margin of safety due to the deletion of this surveillance requirement.

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: Robert A. Capra

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

Date of amendment request: September 29, 1997 (NRC-97-0089)

Description of amendment request: The proposed amendment would relocate the requirements for selected instrumentation and the associated Bases from the technical specifications (TS) to the updated final safety analysis report. The affected instrumentation is seismic monitoring (TS 3.7.2), meteorological monitoring (TS 3.7.3), the traversing in-core probe system (TS 3.7.7), the chlorine detection system (TS 3.7.8), and the loose parts detection system (TS 3.7.10). Changes to the TS index and list of tables were also requested to reflect the relocation of these TS and associated Bases. NRC Generic Letter 95-10, ''Relocation of Selected Technical Specification Requirements Related to Instrumentation,'' dated December 15, 1995, provided information concerning relocation of the requirements for these instruments.

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

The proposed changes would relocate TS 3/4.3.7.2 - Seismic Monitoring Instrumentation, TS 3/4. 3.7.3 - Meteorological Monitoring Instrumentation, TS 3/4.3.7.7 - Traversing In-Core Probe System, TS 3/4.3.7.8 - Chlorine Detection System, and TS 3/4.3.7.10 - Loose-Part Detection System and their associated Bases to the Fermi 2 Updated Final Safety Analysis Report (UFSAR). They would also delete the special reporting requirements from the aforementioned TS which contain such requirements. The proposed changes would revise the TS Index and List of Tables to reflect the relocation of these TS and associated Bases. The relocated TS changes would be controlled in accordance with the requirements of 10 CFR 50.59.

The proposed changes affect TS that do not meet the NRC's ''Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors'' or 10 CFR 50.36(c)(2)(ii) criteria for inclusion in TS. These TS relocations are consistent with NUREG-1433, ''Standard Technical Specifications, General Electric Plants, BWR/4,'' Revision 1, April 1995. Furthermore, these five TS are specifically identified in NRC Generic Letter 95-10, ''Relocation of Selected Technical Specifications Requirements Related to Instrumentation,'' dated December 15, 1995, as suitable for relocation to licensee-controlled documents.

The Special Report requirements of TS 3/4.3.7.2, TS 3/4.3.7.3, and TS 3/4.3.7.10 would be deleted as part of their relocation to the UFSAR. The NRC reporting criteria of 10 CFR 50.72, ''Immediate Notification Requirements for Operating Nuclear Reactors,'' and 10 CFR 50.73, ''Licensee Event Report Systems'' provide appropriate requirements for reporting degraded and non-conforming conditions to the NRC.

These proposed TS changes do not involve a significant increase in the probability of an accident previously evaluated because no changes are being made to any accident initiator. No previously analyzed accident scenario is changed, and initiating conditions and assumptions remain as previously analyzed.

These proposed TS changes do not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not affect accident sequences or assumptions used in evaluating the radiological consequences of an accident. The proposed changes do not alter the source term, containment isolation or allowable radiological releases.

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

The proposed changes do not change the way in which the plant is operated and no new or different failure modes have been defined for any plant system or component. No limiting single failure has been identified as a result of the proposed changes. No new or different types of failures or accident initiators are introduced by the proposed changes.

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

The proposed changes involve instrumentation and systems which are not inputs in the calculation of any safety margin with regard to Technical Specification Safety Limits, Limiting Safety System Settings, Limiting Control Settings or Limiting Conditions for Operation, or other previously defined margins for any structure, system, or component.

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: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226

NRC Project Director: John N. Hannon

Duquesne Light Company, et al., Docket Nos. 50-334 and 50-412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of amendment request: September 11, 1997

Description of amendment request: The proposed amendments would relocate the reactor trip system and engineered safety feature actuation system response times from technical specification (TS) tables 3.3-2 and 3.3-5 to Section 3 of the licensee's Licensing Requirements Manual (LRM) in accordance with the guidance provided in NRC Generic Letter 93-08. Subsequent changes to the LRM would be controlled in accordance with the requirements of 10 CFR 50.59. The proposed amendments would also make several editorial changes in TSs 3.3.1.1 and 3.3.1.2, as well as making conforming changes to the Bases for these TSs.

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

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

The proposed amendment relocates the instrument response time limits for the reactor trip system (RTS) and engineered safety feature actuation system (ESFAS) from the technical specifications to the Licensing Requirements Manual (LRM). The Core Operating Limits Report (COLR) and containment penetrations table (containment isolation valves) are controlled and maintained in the LRM. The LRM was developed to control and maintain those items removed from the technical specifications. The proposed amendment conforms to the guidance given in Enclosures 1 and 2 of Generic Letter 93-08. Neither the response time limits nor the surveillance requirements for performing response time testing will be altered by this submittal. The overall RTS and ESFAS functional capabilities will not be changed and assurance that action requirements of the protective and engineered safety features systems are completed within the time limits assumed in the accident analyses is unaffected by the proposed amendment. Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendment will not change the physical plant or the modes of plant operation defined in the operating license. The change does not involve the addition or modification of equipment nor does it alter the design or operation of plant systems. Therefore, operation of the facility in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The measurement of instrumentation response times at the frequencies specified in the technical specification provides assurance that actions associated with the protective and engineered safety features systems are accomplished within the time limits assumed in the accident analyses. The response time limits, and the measurement frequencies remain unchanged by the proposed amendment. The proposed changes do not alter the basis for any other technical specification that is related to the establishment of or maintenance of a nuclear safety margin. Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001

Attorney for licensee: Jay E. Silberg, Esquire, Shaw, Pittman, Potts &

Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Project Director: John F. Stolz

Entergy Operations, Inc., et al., Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: September 18, 1997

Description of amendment request: The amendment would decrease 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 are ''applicable only for Cycle 10 operation.'' Cycle 10 operation is after the next (i.e., 9th) refueling outage.

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:

I. The proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

The Minimum Critical Power Ratio (MCPR) safety limit is defined in the Bases to Technical Specification [TS] 2.1.1 as that limit which ''ensures that during normal operation and during Anticipated Operational Occurrences (AOOs), at least 99.9% of the fuel rods in the core do not experience transition boiling.'' The MCPR safety limit is re-evaluated for each reload and, for GGNS [Grand Gulf Nuclear Station, Unit 1] Cycle 10, the analyses have concluded that a two-loop MCPR safety limit of 1.11 based on the application of GE's [General Electric Company's] cycle-specific MCPR safety limit methodology is necessary to ensure that this acceptance criterion is satisfied. For single-loop operation, a MCPR safety limit of 1.12 based on GE's cycle-specific MCPR safety limit methodology was determined to be necessary. Core MCPR operating limits are developed to support the Technical Specification [TS] 3.2 requirements and ensure these safety limits are maintained in the event of the worst case transient. Since the MCPR safety limit will be maintained at all times, operation under the proposed changes will ensure [that] at least 99.9% of the fuel rods in the core do not experience transition boiling. Therefore, these changes to the [MCPR] safety limit do not affect the probability or consequences of an accident [previously evaluated].

GE's GESTAR-II approved methodology will continue to be implemented and has no effect on the probability or consequences of any accidents previously evaluated. One exception to GESTAR is that the mis-oriented and mis-located bundle events will continue to be analyzed as accidents subject to the acceptance criteria in the current licensing basis [for GGNS]. The design of the GE11 fuel bundles[, to be added to the core to replace Siemens fuel bundles,] is such that the bundles are not likely to be mis-oriented or mis-located and the normal administrative controls will be in effect for assuring proper orientation and location. Therefore, the probability of a fuel loading error is not increased. This analysis ensures that postulated dose releases will not exceed a small fraction (10 percent) of 10CFR100 [10 CFR Part 100] limits. Therefore, the probability or consequences of accidents previously evaluated are unchanged.

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

The GE 11 fuel to be [added to the core and] used in Cycle 10 [operation] is of a design compatible with fuel present in the core and used in the [current 9th] cycle. [The current core is a mixture of GE11 and Siemens fuel bundles. The addition of GE11 to the core for the 9th cycle is addressed in Amendment 131 to the license dated November 21, 1996.] Therefore, the GE11 fuel will not create the possibility of a new or different kind of accident. The proposed changes do not involve any new modes of operation, any changes to setpoints, or any plant modifications.

They introduce revised MCPR safety limits that have been proven to be acceptable for Cycle 10 operation. Compliance with the applicable criterion for incipient boiling transition continues to be ensured. The proposed MCPR safety limits do not result in the creation of any new precursors to an accident.

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

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

The MCPR safety limits have been evaluated in accordance with GE's current cycle-specific methodology to ensure that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core are not expected to experience transition boiling. Unless otherwise approved, GGNS will implement only the NRC-approved revisions to GE's GESTAR methodology. This GE methodology is similar to those SPC [(Siemens Power Corporation)] reports current listed in TS 5.6.5 and it will be applied in a similar, conservative fashion. [TS 5.6.5, Core Operating Limits Report, lists the analytical methods which are approved by NRC and are used to determine the core operating limits for the GGNS core, including the MCPR.] One exception to GESTAR is that the mis-oriented and mis-located bundle events will continue to be analyzed as accidents subject to the acceptance criteria in the current [GGNS] licensing basis. This analysis ensures that postulated dose releases will not exceed a small fraction (10 percent) of 10CFR100 limits. [The proposed changes are to maintain the margin of safety for transition boiling in the core.] On this basis, the implementation of this GE methodology 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: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., 12th Floor, Washington, DC 20005-3502

NRC Project Director: James W. Clifford, Acting

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

Date of amendment request: September 25, 1997

Description of amendment request: The proposed change modifies Limiting Condition for Operation (LCO) 3.6.1.2 (Containment Leakage), the associated Action, and Surveillance Requirement (SR) 4.6.1.2 in Technical Specification (TS) for Waterford Steam Electric Station, Unit 3 (Waterford 3). The air lock door seal leakage rate acceptance criteria in TS 6.15 is being changed from 0.01La to 0.005La. TS 6.15 is also being modified to make the terms used in the Containment Leakage Rate Testing Program consistent with terms used in the TS. This change corrects an error that inadvertently decreased the allowed outage time from 24 hours to 1 hour when the containment purge valve or containment air lock leakage rates are not within limits. This error was made in the Waterford 3 TS change request that was approved in Amendment 124 for Waterford 3 on April 10, 1997.

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

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

Response: No

The proposed change adds the specific type of containment leakage to the Limiting Condition for Operation (LCO), Action, and Surveillance Requirement (SR) in the Containment Leakage Technical Specification (TS) which results in increasing the allowed outage time from 1 hour to 24 hours when the containment purge valve or containment air lock leakage rates are not within limits. The proposed change revises the air lock door seal leakage rate acceptance criteria. Also, the proposed change revises the Actions in the Containment Leakage TS to be consistent with the Applicability, and revises terms in the Containment Section and Administrative Controls Section of the TS to be consistent with the Containment Leakage Rate Testing Program. This change will not affect the probability of an accident. The containment purge valve and air lock leakage rates are not an initiator of any analyzed event. This change corrects two errors that were made in the Waterford 3 10CFR50 Appendix J, Option B, TS change request that was approved in TS Amendment 124. The first error inadvertently decreased the allowed outage time from 24 hours to 1 hour when either the containment purge valve or containment air lock leakage rate acceptance criteria is not met. The second error inadvertently increased the acceptance criteria for the air lock door seal leakage. The revised air lock door seal leakage rate acceptance criteria was never used at Waterford 3. This change also administratively changes the Containment Leakage TS Action and terms in the TS for consistency.

The proposed change will not affect the consequences of an accident. The amount of leakage from the containment purge valve and from the containment air lock will still be included in the overall combined containment leak rate. Neither the overall containment leakage rate limit nor the Action required to be taken if the overall containment leakage rate were exceeded is being changed. The Containment Leakage TS Action will be consistent with the Applicability and TS 3.0.4 will prohibit entry into Mode 4 (RCS [Reactor Coolant System] temperature ≤ 200°F), unless the overall containment leakage rate is within limit. The revised air lock acceptance criteria was never used. Waterford 3 will continue using the more restrictive acceptance criteria which is controlled administratively. This proposed change does not affect the mitigation capabilities of any component or system, nor does it affect the assumptions relative to the mitigation of accidents or transients.

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

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

Response: No

The proposed change adds the specific type of containment leakage to the LCO, Action, and SR in the Containment Leakage TS. This results in increasing the allowed outage time from 1 hour to 24 hours when the containment purge valve or containment air lock leakage rates are not within limits. The proposed change revises the air lock door seal leakage rate acceptance criteria. Also, the proposed change revises the Actions in the Containment Leakage TS to be consistent with the Applicability, and revises terms in the Containment Section and Administrative Controls Section of the TS to be consistent with the Containment Leakage Rate Testing Program. Neither the design nor configuration of the plant, or how the plant is operated is being changed due to the addition of the specific types of leakage from the Containment Leakage Rate Testing Program, corrections made to the air lock door seal leakage rate acceptance criteria, or the changes made to make the TS consistent. There has been no physical change to plant systems, structures, or components nor will these changes reduce the ability of any of the safety-related equipment required to mitigate anticipated operational occurrences or accidents. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No

The proposed change adds the specific type of containment leakage to the LCO, Action, and SR in the Containment Leakage TS. This results in increasing the allowed outage time from 1 hour to 24 hours when the containment purge valve or containment air lock leakage rates are not within limits. The proposed change revises the air lock door seal leakage rate acceptance criteria. Also, the proposed change revises the Actions in the Containment Leakage TS to be consistent with the Applicability, and revises terms in the Containment Section and Administrative Controls Section of the TS to be consistent with the Containment Leakage Rate Testing Program. The proposed revision to the Action and making the containment leakage rate terms consistent are administrative changes that have no technical impact on the TS.

The pre-amendment 124 Waterford 3 TS and NUREG-1432 allowed entry into specific Actions with allowed outage times greater than 1 hour (24 hours) when the air lock and purge valve leakage rate acceptance criteria could not be met. This change restores this allowed outage time which was inadvertently changed due to an error in the TS change request. The increased allowed outage time may prevent an unnecessary plant shutdown which is a plant transient. Plant shutdowns produce thermal stress on components in the Reactor Coolant System and the potential for a plant upset that could challenge safety systems. This change decreases the possibility of a plant shutdown by replacing the 1 hour allowed outage time with a 24 hour allowed outage time when the containment purge valve or containment air lock leakage is not within limits. Also, the overall containment leakage rate limits are not being changed and are required to be maintained.

The revision to the air lock door seal acceptance criteria is a more restrictive change to correct an error made by Waterford 3 in the TS change request approved in Amendment 124. The less restrictive acceptance criteria was never used; Waterford 3 continued testing to the more restrictive acceptance criteria.

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

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

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

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street N.W., Washington, D.C. 20005-3502

NRC Project Director: James W. Clifford, Acting

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

Date of amendment request: October 1, 1997

Description of amendment request: The proposed amendment would revise the technical specifications (TS) for the Crystal River Nuclear Electric Generating Plant Unit 3 (CR-3). The proposed TS change would add a new TS section, 5.6.2.10.4.c. The new section will provide growth monitoring criteria for the first span section of tubes in the ''B'' Once-Through Steam Generator (OTSG) with pit-like intergranular attack (IGA) indications.

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

Criterion 1

Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The purpose of OTSG tube inspection is to identify tubes that have a higher potential for in service failure due to degradation that results in a reduced ability to withstand normal and upset operating conditions. The formal incorporation of specific indication growth monitoring and repair criteria is consistent with this purpose. Therefore, the probability of an accident previously evaluated has not been increased.

Chapter 14 of the CR-3 Final Safety Analysis Report (FSAR) provides an analysis to assess the consequences of a steam generator tube rupture event, including the complete severance of a steam generator tube. This analyses concluded that CR-3 was sufficiently designed to ensure that in the event of a steam generator tube rupture, the radiological doses would not exceed the allowable limits prescribed by 10 CFR 100. Neither would this result in additional tube failures and further degradation of the integrity of the reactor coolant pressure boundary. The proposed changes do not alter this analysis in any fashion. Therefore, the consequences of an accident have not been increased.

Criterion 2

Does not Create the Possibility of a New or Different Kind of Accident from any Accident Previously Evaluated.

This change does not alter the design or operation of the OTSGs. The incorporation of the proposed requirements is more conservative than the existing ITS requirements. Neither the type of inspection of OTSG tubes nor the process for performing inspections will be changed by this amendment. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3

Does Not Involve a Significant Reduction in the Margin of Safety as defined in the Bases for any Technical Specifications.

The previously performed analyses on the effects of OTSG tube failures, as reported in the CR-3 FSAR, have demonstrated that onsite and offsite consequences are within allowable limits. The proposed change incorporates more conservative growth monitoring and operational assessment criteria for the ''B'' OTSG first-span pit-like IGA indications. This change does not result in a significant reduction in the margin of safety as defined in the Bases for any Technical Specifications.

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

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

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

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: September 26, 1997

Description of amendment request: The proposed amendment would separate the requirements for Control Room Air Conditioning from Control Room Makeup Air and Filtration as presently contained in Technical Specification 3.7.6, ''Control Room Emergency Makeup Air and Filtration,'' and its associated BASES. Technical Specification 3.7.6 now requires that each subsystem of Control Room Emergency Makeup Air and Filtration include an OPERABLE emergency filtration unit and air conditioning unit. The proposed amendment would separate the requirements based on system function. The proposed amendment also would increase the allowed outage time for the air conditioning portion of the Control Room Air Conditioning Subsystem.

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

A. The changes do not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)). The proposed changes have no impact on the probability of an accident because the control room ventilation systems are support systems which have a role in the detection and mitigation of accidents but do not contribute to the initiation of any accident previously evaluated. Reorganizing the Technical Specifications by function is merely an administrative change and the change has no impact on the course of any accidents previously evaluated since there is no change in the functions provided by the subsystems.

Increasing the allowed outage time to 30 days from 7 days for the cooling of recirculated air while one train is inoperable does not affect the availability of the second train of air conditioning or the actions required if both trains of air conditioning become unavailable. Thus, the consequences accidents previously evaluated are not increased.

B. The changes do not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because they do not affect the function of any facility structure, system or component, nor do they affect the manner by which the facility is operated. The proposed changes do not introduce any new failure modes.

C. The changes do not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)) because the proposed changes do not affect the function of any facility structure, system or component, nor do they affect the manner by which the facility is operated. Increasing the allowed outage time for the cooling of recirculated air while one train is inoperable represents an increase in the probability that the air conditioning functions could be unavailable. However, the increase does not affect the availability of the second train of air conditioning or the actions required should both trains of air conditioning become unavailable.

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: Exeter Public Library, Founders Park, Exeter, NH 03833

Attorney for licensee: Lillian M. Cuoco, Esquire, Northeast Utilities Service Company, Post Office Box 270, Hartford CT 06141-0270

NRC Project Director: Ronald B. Eaton, Acting

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

Date of amendment requests: September 26, 1997

Description of amendment requests: The proposed amendments would revise Technical Specification (TS) 3.4.B, ''Auxiliary Feedwater System,'' to provide specific guidance for conducting post-maintenance operational testing of the turbine-driven auxiliary feedwater (TDAFW) pump and associated system valves to meet operability and limiting conditions for operation during unit startup. An additional change is proposed to revise Table TS.3.5.2B to permit during Mode 2 the bypassing of the auto start feature of the auxiliary feedwater (AFW) pumps that results from the trip of both main feedwater pumps when the feedwater pumps are not required to be operated.

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.

Since none of the proposed changes involve a physical change to the plant, the mechanisms that could cause a Loss of Normal Feedwater have not changed. The probability that a Loss of Normal Feedwater will occur is not altered.

This change still requires that the motor driven AFW Pump and associated system valves are operable during Startup Operations. Analysis of the Loss of Normal Feedwater transient shows that a single AFW Pump provides sufficient AFW flow to prevent any adverse conditions in the core. The condition of an inoperable TDAFW Pump is already permitted during power operations where the consequences of the event would be more severe than during startup. Since there are no consequences from the Loss of Normal Feedwater event at power, the consequences during startup would still be none, but the margins would be larger because; (1) the amount of residual heat generated is less because reactor power at the start of the event is less and (2) the power history is lower resulting in less decay heat.

Thus, these changes do not involve an increase in the probability or consequences of an accident previously analyzed.

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 changes do not create the possibility of a new or different kind of accident previously evaluated because the proposed changes do not introduce a new mode of operation or testing, or make physical changes to the plant.

The proposed changes do not alter the design, function, operation, or testing of any plant component, therefore the possibility of a new or different kind of accident from those previously analyzed would not be created by these changes to Technical Specifications.

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

Margins previously established for the Loss of Normal Feedwater event, were analyzed for different initial conditions. The Loss of Normal Feedwater event was analyzed for Power Operations. This analysis determined that no adverse conditions would occur in the core. Since there are no consequences from the Loss of Normal Feedwater event at power, the consequences during startup would still be none but the margins would be greater because; (1) the amount of residual heat generated is less because reactor power at the start of the event is less and (2) the power history is lower causing less decay heat.

Therefore, the proposed change does not result in a significant reduction in the margin of safety currently established.

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: John N. Hannon

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

Date of amendment request: September 3, 1997

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) to revise the number of hours operating personnel can work in a normal shift. The proposed amendment also contains some administrative changes to the TSs.

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

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

A. Establishing operating personnel work hours at, ''an 8 to 12 hour day, nominal 40 hour week,'' allows normal plant operations to be managed more effectively and does not adversely effect performance of operating personnel. Overtime remains controlled by site administrative procedures in accordance with NRC Policy Statement on working hours (Generic Letter 82-12). If 8 hour shifts are maintained in part or whole, then acceptable levels of performance from operating personnel is assured through effective control of shift turnovers and plant activities. No physical plant modifications are involved and none of the precursors of previously evaluated accidents are affected. Therefore, this change will not involve a significant increase in the probability or consequence of an accident previously evaluated.

B. Editorial changes clarify section 6.2.2.g without changing the intent or meaning. The proposed change meets the intent of the NRC Policy Statement on working hours (Generic Letter 82-12).

C. Changes to sections 3.10.6.1.a and 3.10.9 do not change the intent or meaning of the technical specification sections. Clarification to the table notation in section 4.1 related to the definition of shift checks to monitor plant conditions will continue as intended but are allowed to increase up to at least once per 12 hours. This increase is consistent with standard industry practice as represented by the Standard Technical Specifications (STS), Reference 1.

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

A. Establishing operating personnel work hours at, ''an 8 to 12 hour day, nominal 40 hour week,'' allows normal plant operations to be managed more effectively and does not adversely effect performance of operating personnel. If 8 hour shifts are maintained in part or whole, then acceptable levels of performance from operating personnel is assured through effective control of shift turnovers and plant activities. Overtime remains controlled by site administrative procedures in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12). No physical modification of the plant is involved. As such, the change does not introduce any new failure modes or conditions that may create a new or different accident. Therefore, operation in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

B. Editorial changes clarify section 6.2.2.g without changing the intent or meaning. The proposed change meets the intent of the NRC Policy Statement on working hours (Generic Letter 82-12).

C. Changes to sections 3.10.6.1.a and 3.10.9 do not change the intent or meaning of the technical specification sections. Clarification to the table notation in section 4.1 related to the definition of shift checks to monitor plant conditions will continue as intended but are allowed to increase up to at least once per 12 hours. This increase is consistent with standard industry practice as represented by the Standard Technical Specifications (STS), Reference 1.

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

A. Establishing operating personnel work hours at, ''an 8 to 12 hour day, nominal 40 hour week,'' allows normal plant operations to be managed more effectively and does not adversely effect performance of operating personnel. If 8 hour shifts are maintained in part or whole, then acceptable levels of performance from operating personnel is assured through effective control of shift turnovers and plant activities. Overtime remains controlled by site administrative procedures in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12) and is consistent with the Standard Technical Specifications. The proposed change involves no physical modification of the plant, or alterations to any accident or transient analysis. There is no Basis to section 6 of the Technical Specifications, and the changes are administrative in nature. Therefore, the change does not involve any significant reduction in a margin of safety.

B. Editorial changes clarify section 6.2.2.g without changing the intent or meaning. The proposed change meets the intent of the NRC Policy Statement on working hours (Generic Letter 82-12).

C. Changes to sections 3.10.6.1.a and 3.10.9 do not change the intent or meaning of the technical specification sections. Clarification to the table notation in section 4.1 related to the definition of shift checks to monitor plant conditions will continue as intended but are allowed to increase up to at least once per 12 hours. This increase is consistent with standard industry practice as represented by the Standard Technical Specifications (STS), Reference 1.

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

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

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

NRC Project Director: S. Singh Bajwa, Director

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

Date of amendment request: September 8, 1997

Description of amendment request: The proposed amendment would revise the f delta I function.

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

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

Response:

No. The revision to the negative [f delta I] penalty does not significantly increase the probability or consequences of an accident previously evaluated in the FSAR [Final Safety Analysis Report]. This revision does not directly initiate an accident. The consequences of accidents previously evaluated in the FSAR are unaffected by this proposed change because no change to any equipment response or accident mitigation scenario has resulted. There are no additional challenges to fission product barrier integrity.

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

Response:

No. The revision to the negative [f delta I] penalty does not create the possibility of a new or different kind of accident than any accident already evaluated in the FSAR. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of this proposed change. The proposed Technical Specification revision does not challenge the performance or integrity of any safety related systems. Therefore, the possibility of a new or different kind of accident is not created.

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

Response:

No. The proposed change to the Technical Specification does not involve a significant reduction in a margin of safety. The margin of safety associated with the acceptance criteria for any accident is unchanged.

The revision to the negative [f delta I] penalty will have no affect on the availability, operability or performance of the safety related systems and components and does not affect the plant Technical Specification requirements. The revision to the negative [f delta I] penalty does require a change to the Technical Specifications but does not prevent inspections or surveillances required by the Technical Specifications.

In addition, the revision to the [f delta I] parameters is based upon the revised boron dilution rate used to analyze the boron dilution transient. Indian Point 3 procedures require the placement of one PW [primary water makeup] pump control switch in the pull-out position, thus ensuring that only one PW pump is operating.

The Bases of the Technical Specifications are founded in part on the ability of the regulatory criteria being satisfied assuming the limiting conditions for operation for various systems. Conformance to the regulatory criteria for operation with the revision to the negative [f delta I] penalty is demonstrated and the regulatory limits are not exceeded. Therefore, the margin of safety as defined in the Technical Specifications is not reduced.

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

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

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

NRC Project Director: S. Singh Bajwa, Director

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

Date of amendment request: September 29, 1997

Description of amendment request: The proposed amendment would revise the Ginna Station Improved Technical Specifications (ITS) to change the Allowable Value for high steam flow input into limiting condition for operation (LCO) Table 3.3.2-1, Function 4.d.

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 Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. An increase in the high steam flow Allowable Value for LCO Table 3.3.2-1, Function 4.d does not increase the probability of any analyzed accident nor does it increase the likelihood of an inadvertent main steam isolation. This function is not explicitly credited in the accident analyses. Also, there are three coincident parameters which must be reached in order for this function to cause a main steam line isolation. It has been demonstrated that the change to the high steam flow parameter does not delay the time at which this isolation signal would be reached for any analyzed accident since the steam flow value is reached much earlier in the accident scenario than the other parameters. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes do not directly affect any analyzed accident analysis. The new isolation times will not be affected for analyzed accidents. As such, no question of safety is involved, and the 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: Rochester Public Library, 115 South Avenue, Rochester, New York 14610

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

NRC Project Director: S. Singh Bajwa, Director

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

Date of amendment request: August 26, 1997

Description of amendment request: The proposed amendment would change Technical Specification (TS) 3/4.6.1.3, ''Containment Systems - Containment Air Locks,'' TS Bases 3/4.6.1.3, ''Containment Systems - Containment Air Locks,'' and TS Bases 3/4.9.4, ''Refueling Operations - Containment Penetrations.'' The containment air lock Limiting Condition for Operation and Surveillance Requirements would be modified, and the associated bases would be changed.

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 Davis-Besse Nuclear Power Station has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because accident initiators, conditions, or assumptions are not affected by the proposed changes, which clarify the Technical Specification (TS) Limiting Condition for Operation (LCO) for the containment air locks, extend the test frequency for the containment air lock interlock mechanisms, and modify guidelines relative to the routing of hoses and cables through the containment air lock during core alterations or during movement of irradiated fuel within the containment.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not change the source term, containment isolation, or allowable releases. The proposed changes do not affect the allowable containment leakage rates presently specified in the Technical Specifications.

The proposed change to Surveillance Requirement (SR) 4.6.1.3.c to increase the surveillance interval for the air lock interlock mechanism to ''at least once per REFUELING INTERVAL'' is justified due to the purely mechanical nature of the interlock mechanism, and given that the interlock mechanism is not normally challenged when the air lock door is used for entry and exit since administrative controls require strict adherence to single door opening. Operating experience shows that the interlock mechanisms are very reliable. Further, the proposed change will allow performance of the surveillance under the conditions that apply during a plant outage, which is preferable to performance, in part, with the plant at power, as is currently necessitated by the present six month interval surveillance requirement. Although an interlock mechanism failure would not affect air lock sealing capabilities and would therefore not directly affect containment integrity, performance of the surveillance with the plant at power, when containment integrity is required, carries with it the potential for loss of containment integrity, should the interlock fail during testing and allow both doors to be opened simultaneously. The proposed TS change may result in an increased probability that due to the increased [decreased] test frequency, an inoperable interlock mechanism could go undetected for a longer length of time. However, in the unlikely event that as a containment entry is being made, abnormal radiation levels inside containment occur, any increase in consequences due to a radioactive release as a result of an inadvertent opening of both air lock doors (as could be allowed by a failed interlock mechanism and assuming violation of administrative controls) is counter-balanced by the decreased likelihood of similar events occurring when the interlock mechanism is tested at power under the current, more frequent, test requirement.

The proposed change to TS Bases 3/4.9.4 to add flexibility in routing cable and hoses through the containment personnel air lock will not affect the requirement to maintain at least one containment personnel air lock door capable of being closed. The analysis results for a fuel handling accident inside containment, as presented in Section 15.4.7.3 of the DBNPS Updated Safety Analysis Report (USAR), are well within the 10 CFR 100 guideline values. Since the analysis does not take credit for containment isolation, the status of the personnel air lock has no impact on the acceptability of the results. Under the proposed change, in the event of a fuel handling accident, release of radioactive material will continue to be minimized since at least one personnel air lock door will remain capable of being closed.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes. The proposed changes do not involve a change to the plant design or operation and, therefore, will not introduce any new or different failure modes or initiators.

3. Not involve a significant reduction in a margin of safety.

The proposed TS change to SR 4.6.1.3.c to increase the surveillance interval for the air lock interlock mechanism will have no adverse effect on plant safety based on its good historical surveillance and maintenance data, and the reduction in testing at power which will occur.

The analysis results for a fuel handling accident inside containment, as presented in the D

Basis for proposed no significant hazards guideline values. Since the analysis does not take credit for containment isolation, the status of the personnel air lock has no impact on the acceptability of the results. Therefore, the proposed change to TS Bases 3/4.9.4 to add flexibility in routing cable and hoses through the containment personnel air lock will not reduce the margin of safety.

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

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

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: October 11, 1996

Description of amendment request: The proposed ammendment would revise the Vermont Yankee Technical Specifications (TSs) regarding the amount of foam concentrate required to support operability of the Recirculation Motor Generator (M. G.) Set Foam System as stated in TS 3.13.G.1 and 3.13.G.2. In both instances, the required amount of foam concentrate would be increased from 100 to 150 gallons.

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

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

The changes proposed herein affect only the amount of foam concentrate inventory required to support the operability of the Recirculation M. G. Set Foam System and therefore does not modify or add any initiating parameters that would significantly increase the probability or consequences of any previously analyzed accident.

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

These changes involve the upgrade of an existing system using standard fire protection components to provide the level of protection originally required. An evaluation has been completed to ensure that the enhanced spray pattern and increased volume of spray does not impact any equipment not previously evaluated and does not create any threat of flooding to equipment. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

These changes do not affect any equipment involved in potential initiating events or safety limits. Therefore, it is concluded that 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: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301

Attorney for licensee: Mr. David R. Lewis, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, DC 20037-1128

NRC Project Director: Ronald B. Eaton, Acting Director

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

Date of amendment request: September 2, 1997

Description of amendment request: This license amendment request proposes to revise Technical Specification 3.7.1.2, Auxiliary Feedwater System, and associated Bases, to add requirements for the essential service water (ESW) flowpaths to the turbine-driven auxiliary feedwater pump (TDAFWP) and other changes consistent with the technical specification conversion application previously submitted. The proposed revisions would (a) provide an action and allowed outage time (AOT) for inoperability of one of the redundant ESW flowpaths to the TDAFWP, and (b) incorporate an action and AOT for inoperability of one of the redundant steam flowpaths to the TDAFWP turbine and other changes to make the auxiliary feedwater system limiting condition for operation (LCO) and actions consistent with those previously submitted.

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.

ESW Flow Path Required Actions

This change would provide a 7-day AOT for the ESW supply flow paths to the TDAFWP. This would replace administrative controls that imposed a 72-hour AOT on ESW flow paths to the TDAFWP.

The proposed change does not result in any hardware changes or changes to operating methodologies. This revision does not affect an accident initiator of any analyzed accident since the TDAFWP ESW supply only provides flow to equipment required to mitigate the consequences of an accident. The revision recognizes that the TDAFWP would remain available in most cases for accident mitigation because of the low probability of an accident and subsequent equipment failure requiring the use of the inoperable ESW supply for the TDAFWP. Changing the AOT from 3 days to 7 days would have a negligible effect on this small probability. Loss of the AFW function would also require the failure of the MDAFWPs [motor-driven auxiliary feedwater pumps]. In addition, the CST [condensate storage tank] would be OPERABLE in accordance with LCO 3.7.1.3 and would be available for use by the TDAFWP for all events except those external hazards that represent a hazard to the integrity of the tank itself.

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

2. Steam Supply Flow Path Required Actions

This change would provide a 7-day AOT for the steam supply flow paths to the TDAFWP. This would replace an administrative control that required the TDAFWP to be declared inoperable without applying an AOT. The proposed change does not result in any hardware changes or changes to operating methodologies. This revision does not affect an accident initiator of any analyzed accident since the TDAFWP steam supply only provides power to equipment required to mitigate the consequences of an accident. The revision recognizes the low probability of an accident requiring the use of the inoperable steam supply for the TDAFWP coincident with the failure of the MDAFWPs.

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

3. Use of ''Trains'' Instead of ''Pumps and Associated Flow Paths'' and Removal of Unnecessary Details

This change is partially administrative and partially a movement of provisions not required to be in the technical specifications to other controlled documents. The administrative change does not impact initiators of analyzed events or equipment assumed in the mitigation of accidents or transient events. The details moved from the technical specification would be located in the Bases of the technical specification. Since any changes to the Bases will be evaluated per the requirements of 10 CFR 50.59, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

4. Twelve Hours to HOT SHUTDOWN

This change would allow an additional 6 hours to achieve HOT SHUTDOWN for the AFW System. The proposed change does not alter the plant configuration or operation or function of any safety system. Consequently, the change does not increase the probability of an accident as defined in accident analysis. The proposed change permits a longer time to cooldown to RHR [residual heat removal] entry conditions; however, this would not affect the consequences of any postulated accidents and is appropriate due to the need to avoid any transients while cooling down with a potentially degraded AFW System.

Therefore, the proposed change would have no significant effect on the probability or consequences of any previously analyzed accidents.

5. Additional AOT of 10 Days from Discovery of Failure to Meet the LCO

The proposed change imposes more stringent requirements than contained in current technical specification. The more stringent requirements are imposed to ensure that the OPERABILITY requirements for the AFW System are maintained consistent with the safety analysis and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

6. Suspension of LCO 3.0.3

The proposed change involves clarifying the technical specification. The proposed revision involves no technical changes to the current technical specification. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

1.ESW Flow Path Required Actions

The proposed change to add a 7-day AOT for the ESW supply flow paths does not require physical alteration to any plant system or change the method by which any safety-related system performs its function.

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

2. Steam Supply Flow Path Required Actions

The proposed change to add a 7-day AOT for the steam supply flow paths does not require physical alteration to any plant system or change the method by which any safety-related system performs it function.

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

3. Use of ''Trains'' Instead of ''Pumps and Associated Flow Paths'' and Moving of Unnecessary Details

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in controlling parameters. The proposed change will not impose any different requirements and adequate control of the information moved to the Bases will be maintained. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

4. Twelve Hours to HOT SHUTDOWN

The proposed change does not require physical alteration to any plant system or change the method by which any safety-related system performs its function. As discussed above, the change does allow additional time to complete transfer from the SG [steam generator] as the method for heat removal to the RHR System, but does not alter the basic methodology.

Therefore, the proposed change would not create the possibility of a new or different kind of accident.

5. Additional AOT of 10 Days from Discovery of Failure to Meet the LCO

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in controlling parameters. The proposed change does impose different (more restrictive) requirements. However, these changes remain consistent with assumptions made in the safety analysis regarding system OPERABILITY. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

6. Suspension of LCO 3.0.3

The proposed change clarifies an implied requirement from current technical specifications and does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in controlling parameters. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

1. ESW Flow Path Required Actions

The proposed change to add a 7-day AOT for the ESW flow paths does not change any accident analysis assumptions, initial conditions or results. Consequently, it does not have an effect on margin of safety.

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

2. Steam Supply Flow Path Required Actions

The proposed change to add a 7-day AOT for the steam supply flow paths does not change any accident analysis assumptions, initial conditions or results. Consequently, it does not have an effect on margin of safety.

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

3. Use of ''Trains'' Instead of ''Pumps and Associated Flow Paths'' and Removal of Unnecessary Details

The proposed change will not reduce a margin of safety because it has no impact on the design basis or safety analysis. In addition, the requirements to be transposed from the technical specification to the Bases are the same as the current technical specification. Since any future changes to these requirements in the Bases will be evaluated per the requirements of 10 CFR 50.59, proper controls are in place to maintain an appropriate margin of safety. Therefore, the changes do not involve a significant reduction in a margin of safety.

4. Twelve Hours to HOT SHUTDOWN

The proposed change does not alter the basic regulatory requirements or change any accident analysis assumptions, initial conditions or results.

Therefore, the proposed change would have no significant adverse effect on margins of safety.

5. Additional AOT of 10 Days from Discovery of Failure to Meet the LCO

The imposition of more stringent requirements on AOT would increase the margin of plant safety by providing additional requirements to maintain AFW System OPERABILITY.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

6. Suspension of LCO 3.0.3

The proposed change will not reduce a margin of safety because it has no impact on the design basis or safety analysis. This change is administrative in nature. As such, no question of safety is involved.

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

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037

NRC Project Director: William H. Bateman

Yankee Atomic Electric Company, Docket No. 50-029, Yankee Nuclear Power Station, Franklin County, Massachusetts

Date of amendment request: September 5, 1997 (Accession No. 9709100106)

Description of amendment request: The proposed technical specification (TS) changes are needed to permit removal of spent nuclear fuel from the Spent Fuel Pit storage racks into a combined storage/shipping cask and to enable handling of the cask components and other hardware by the Yard Area Crane. Specific TS changes are needed for minimum water coverage over spent fuel, shielding for personnel exposure, increased loads carried over the fuel, addition of restrictions for load paths over spent fuel and changes to the appropriate TS bases.

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

The changes provide for an alternate method of providing protection of the spent fuel and spent fuel pit (SFP) from heavy loads that must be transported over the SFP. The method chosen, that is, providing a single-failure-proof overhead crane, is considered an acceptable method as stated in Regulatory Guide 1.13, ''Spent Fuel Storage Facility Design Basis,'' and NUREG-0612, ''Control of Heavy Loads at Nuclear Power Plants.'' The Defueled Technical Specification 3.1.2 requirement for five (5) feet of water above the top of the fuel assemblies for fuel traveling in the SFP is provided for personnel protection (ALARA). This protection is provided by the shielding afforded by the shipping and/or transfer cask system. The cask handling crane will comply with the single-failure-proof crane design requirements of NUREG-0554, ''Single Failure-Proof Cranes for Nuclear Power Plants,'' and meet the criteria specified in NUREG-0612. In addition, design controls and administrative controls will be maintained to prevent handling of the shipping and/or transfer cask over spent fuel in the SFP. As such, these changes will not:

1. Involve a significant increase in the probability or consequence of an accident previously evaluated. NUREG-0612, Section 5, provides direction for providing an adequate level of defense-in-depth for handling of heavy loads near spent fuel and safe shutdown systems. The single-failure-proof overhead crane design is presented as an acceptable method of providing the proper margin of safety for handling of heavy loads. By upgrading the cask handling crane to a single-failure-proof design and meeting the requirements presented in Sections 5.1.1 and 5.1.6 of NUREG-0612 (for safe load path, procedures, crane operator training and qualification, special lifting devices, lifting devices that are not specially designed, and crane inspection, testing, and maintenance) a sufficient level of defense-in-depth is provided to ensure that a load drop is not a credible event. As such, there is no increase in the probability or consequence of an accident previously evaluated as a result of the heavy load changes. A fuel handling incident is a currently analyzed event; dropping of a fuel assembly over the spent fuel within the transfer cask is similar to dropping of a fuel assembly over spent fuel in the SFP. The design basis fuel handling event analysis bounds these events, so there is no increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated. The defense-in-depth philosophy provided by the single-failure-proof crane load handling sysem design, and compliance with the requirements specified in Sections 5.1.1 and 5.1.6 of NUREG-0612 provide assurance that for a credible single failure of the crane load handling system, the system will still be able to perform its safety function. This provides assurance that a load drop accident is not a credible event. As such, no new or different kind of accident will be created from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety. The proposed changes implement the guidelines of NUREG-0612 and Regulatory Guide 1.13. YAEC is implementing an acceptable alternate method of ensuring the safe handling of heavy loads

over the SFP. This method provides a defense-in-depth approach for handling of heavy loads over the SFP and maintains the margin of safety consistent with that of the current requirements. Further protection is provided by the prohibition of these additional heavy loads from travel over the spent fuel assemblies in the SFP racks. The use of a single-failure-proof crane and associaed lifting devices provide an increased margin of safety that ensure that a load drop event is not credible and is considered an adequate alternate for the additional area added to the safe load path. The use of a limit switch to prevent movement of the prohibited cask handling crane loads from movement beyond the safe load path, provides an additional margin of safety, that was previously provided by the steel framing at the southern edge of the SFP superstructure roof opening. The single-failure-proof crane and defense-in-depth design ensure that a load drop is not a credible event, assuring that the margin of safety is not reduced.

Based on the above considerations, it is concluded that there is reasonable assurance that the operation of Yankee Nuclear Power Station consisent with the proposed changes will not endanger the health and safety of the public.

The proposed change has been reviewed by the Plant Operations Review Committee and the Nuclear Safety Audit and Review Committee.

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: Greenfield Community College, 1 College Drive, Greenfield, Massachusetts 01301

Attorney for licensee: Thomas Dignan, Esquire, Ropes and Gray, One International Place, Boston, Massachusetts 02110-2624

NRC Project Director: Seymour H. Weiss

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.

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: September 12, 1997

Brief description of amendment: The proposed amendment involves a revision to the Emergency Diesel Generator protective relaying scheme at CR3, as described in the Final Safety Analysis Report Chapter 8.

Date of publication of individual notice in the **Federal Register:** September 30, 1997 (62 FR 51165).

Expiration date of individual notice: October 30, 1997

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

Notice Of Issuance Of Amendments To Facility Operating Licenses

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

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

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

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

Arizona Public Service Company, et al., Docket Nos. STN 50-528, STN 50-529, and STN 50-530, Palo Verde Nuclear Generating Station, Units Nos. 1, 2, and 3, Maricopa County, Arizona

Date of application for amendment: March 24, 1995, as supplemented by letters dated September 10, 1995, and March 22, 1996.

Brief description of amendment: The amendment would change the technical specifications (TS) to (1) reflect the applicable portions of NUREG-1432, ''Standard Technical Specifications Combustion Engineering Plants,'' (2) implement the recommendations of Generic Letter (GL) 93-05, ''Line Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Plant Operation,'' and (3) implement the recommendations of GL 94-01, ''Removal of Accelerated Testing and Specific Reporting Requirements for Emergency Diesel Generators.'' The purpose of the proposed amendment is to increase emergency diesel generator (EDG) reliability by reducing stresses on EDG caused by unnecessary testing. The associated Bases are also updated.

Date of issuance: October 6, 1997

Effective date: October 6, 1997, to be implemented within 120 days of date of issuance.

Amendment Nos.: Unit 1 - 114; Unit 2 - 107; Unit 3 - 86

Facility Operating License Nos. NPF-41, NPF-51, and NPF-74: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 6, 1995 (60 FR 29870) The September 10, 1995, and March 22, 1996, supplemental letters provided additional clarifying information and did not change the original no significant hazards consideration. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 6, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004

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: March 28, 1996, as supplemented November 20, 1996, and July 31, 1997.

Brief description of amendments: The amendments reduce the moderator temperature coefficient limit shown on Technical Specification Figure 3.1.1-1. This proposed change is necessary to support changes in the safety analyses made to accommodate a larger number of plugged steam generator tubes for future operating cycles.

Date of issuance: October 2, 1997

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

Amendment Nos.: 222 and 198

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

Date of initial notice in **Federal Registe for amendment: February 21, 1997**

Brief description of amendment: This amendment adds a specific time limit to Technical Specification Table 3.3-3 to place an inoperable refueling water storage tank level channel in a bypassed condition.

Date of issuance: September 30, 1997

Effective date: September 30, 1997

Amendment No.: 74

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

Date of initial notice in **Federal Register:** April 9, 1997 (62 FR 17225) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 30, 1997. No significant hazards consideration comments received: No.

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

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: July 1, 1997

Brief description of amendments: The amendments revise Technical Specification Table 3.3.7.1-1, ''Radiation Monitoring Instrumentation,'' to require two channels to be operable per trip system as opposed to two per intake. This change reflects a modification to the design of the instrumentation logic to satisfy single failure requirements. The amendments also revise the associated action statement to clarify system logic wording.

Date of issuance: October 9, 1997

Effective date: Immediately, to be implemented within 60 days.

Amendment Nos.: 121 and 106

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

Date of initial notice in **Federal Register:** August 27, 1997 (62 FR 45455). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 9, 1997. No significant hazards consideration comments received: No.

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

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 application for amendments: May 1, 1997

Brief description of amendments: The amendments clarify the load value for the emergency diesel generator to be equal to or greater than the largest single load and revise the frequency and voltage requirements during the performance of the test.

Date of issuance: October 7, 1997

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

Amendment Nos.: 178 and 176

Facility Operating License Nos. DPR-29 and DPR-30: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 18, 1997 (62 FR 33121). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 7, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: January 10, 1996, as supplemented February 20, 1997

Brief description of amendment: The amendment revises the Technical Specifications for the containment emergency escape air lock test requirements. Concurrently, the Commission has also granted an exemption to certain requirements of 10 CFR Part 50, Appendix J, relating to the testing of the emergency escape air lock, to the extent that leakage rate testing is not necessary after opening the emergency escape air lock doors for post-test restoration or seal adjustment.

Date of issuance: September 30, 1997

Effective date: September 30, 1997

Amendment No.: 177

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

Date of initial notice in **Federal Register:** February 26, 1997 (62 FR 8795) The February 20, 1997, letter provided clarifying information within the scope of the original application and did not change the NRC staff's initial proposed no significant hazards considerations determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 30, 1997. 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 6, 1995, as supplemented October 18 1996, January 10 and June 27, 1997

Brief description of amendment: The amendment deletes crane operation and movement of heavy loads requirements and their bases from the technical specifications. The requirements have been incorporated into the Palisades Operating Requirements Manual (ORM). The ORM has been incorporated by reference into the Palisades Final Safety Analysis Report, assuring that future changes to the crane and heavy loads requirements will be subject to the provisions of 10 CFR 50.59.

Date of issuance: October 2, 1997

Effective date: October 2, 1997

Amendment No.: 178

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

Date of initial notice in **Federal Register:** July 17, 1996 (61 FR 37298) The October 18, 1996, January 10 and June 27, 1997, letters provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 1997. No significant hazards consideration comments received: No.

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

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

Date of amendment request: September 5, 1997 (NRC-97-0107)

Description of amendment request: The amendment revises the Technical Specifications by adding a special test exception to allow reactor coolant temperatures up to 212 degrees Fahrenheit during hydrostatic or inservice leak testing while in Operational Condition 4 without entering Operational Condition 3. The amendment also makes related changes to the Index, Table 1.2, ''Operational Conditions,'' and the Bases to incorporate the reference to the proposed special test exception. Date of issuance: September 30, 1997

Effective date: September 30, 1997, with full implementation within 45 days

Amendment No.: 114

Facility Operating License No. NPF-43: Amendment revises the Technical Specifications and Bases.

Date of initial notice in **Federal Register:** September 30, 1997 (62 FR The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of Michigan, and final determination of no significant hazards considerations are contained in a Safety Evaluation dated September 30, 1997 No significant hazards consideration comments received: No.

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

Duke Energy Corporation, et al., Docket No. 50-413, Catawba Nuclear Station, Unit 1, York County, South Carolina

Date of application for amendment: May 8, 1997, as supplemented by letter dated September 10, 1997

Brief description of amendment: The amendment revises Section 3/4.1.2 of the Technical Specifications to permit a one-time natural circulation test during Mode 3.

Date of issuance: October 9, 1997

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

Amendment No.: 162

Facility Operating License No. NPF-35: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 4, 1997 (62 FR 30631) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 9, 1997. No significant hazards consideration comments received: NoLocal Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

Entergy Gulf States, Inc., Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: August 5, 1997, as supplemented August 15, 1997

Brief description of amendment: The amendment revises the Technical Specifications to increase the two recirculation loop Minimum Critical Power Ratio (MCPR) safety limit to 1.13 and the single recirculation loop MCPR safety limit to 1.14.

Date of issuance: October 8, 1997

Effective date: October 8, 1997

Amendment No.: 99

Facility Operating License No. NPF-47: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 27, 1997 (62 FR 45456) The August 15, 1997, submittal provided clarifying information that did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 8, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 1, 1997

Brief description of amendment: Revises the Technical Specifications (TS) to extend the surveillance interval for the Engineered Safety Features Actuation System to a refueling interval on a staggered test basis.

Date of Issuance: October 2, 1997

Effective Date: October 2, 1997

Amendment No.: 90

Facility Operating License No. NPF-16: Amendment revised the TS.

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 October 2, 1997. 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

GPU Nuclear Corporation, et al., Docket No. 50-289, Three Mile Island Nuclear Station, Unit No. 1 (TMI-1), Dauphin County, Pennsylvania

Date of application for amendment: August 14, 1997, as supplemented September 9, 19, and 24, 1997

Brief description of amendment: The amendment revises the TMI-1 Technical Specifications which decreases the maximum allowable dose equivalent iodine-131 limit in the reactor primary coolant from 1.0 uCi/gm to 0.35 uCi/gm.

Date of Issuance: October 2, 1997

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

Amendment No.: 204

Facility Operating License No. NPF-50: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 27, 1997 (62 FR 45459) The supplemental letters did not affect the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 10, 1997

Brief description of amendment: The amendment changes the Technical Specifications (TSs) by relocating the TS surveillance requirement for attaining a negative pressure in the enclosure building, addressing operability, deleting the definition for enclosure building integrity, modifying enclosure building access opening requirements, and making editorial changes for clarification and consistency. The TS Bases are also updated to reflect the proposed changes including the need to maintain the integrity of the enclosure building and to support previously approved laboratory testing requirements for charcoal filter sample testing.

Date of issuance: September 30, 1997

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

Amendment No.: 208

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

Date of initial notice in **Federal Register:** May 7, 1997 (62 FR 24987) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 30, 1997. No significant hazards consideration comments received: No

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385

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

Date of application for amendment: July 18, 1997

Brief description of amendment: The amendment adds a new Technical Specification and associated Bases to address the operability of the steam generator atmospheric relief bypass valves.

Date of issuance: October 2, 1997

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

Amendment No.: 151

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

Date of initial notice in **Federal Register:** August 13, 1997 (62 FR 43370) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut 06385

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50-278, Peach Bottom Atomic Power Station, Unit No. 3, York County, Pennsylvania

Date of application for amendment: January 17, 1995, as supplemented by letters dated March 30, 1995, July 2, 1996, February 28, 1997, and September 22, 1997

Brief description of amendment: The amendment revised the technical specifications to support the replacement of the Source Range and Intermediate Range Monitors with the Wide Range Neutron Monitoring System.

Date of issuance: September 30, 1997

Effective date: As of its date of issuance and is to be implemented upon completion of Unit 3 Modification P00271.

Amendment No.: 224

Facility Operating License No. DPR-56: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 6, 1995 (62 FR 29885) The March 30, 1995, July 2, 1996, February 28, 1997, and September 22, 1997, supplemental letters 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 September 30, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 14, 1997

Brief description of amendment: The amendment revises Appendix A, Section 6 of the James A. FitzPatrick Technical Specifications. These changes will enable the Safety Review Committee to review rather than audit plant staff performance by deleting the plant staff performance audit requirements from Section 6.5.2.9.b and incorporating a plant staff performance review requirement in Section 6.5.2.8. Additionally, this amendment application replaces the position title of Vice President Regulatory Affairs and Special Projects with Director Regulatory Affairs and Special Projects.

Date of issuance: October 3, 1997

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

Amendment No.: 240

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

Date of initial notice in **Federal Register:** August 13, 1997 (62 FR 43374) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 3, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 19, 1997, as supplemented by letters dated July 30 and 31, 1997

Brief description of amendment: This amendment changes TS 4.1.3.1.2, ''Control Rod Operability;'' TS 3.1.3.6, ''Control Rod Drive Coupling;'' TS 3.1.3.7, ''Control Rod Position Indication;'' TS 3.1.4.1, ''Rod Worth Minimizer;'' TS 3/4.1.4.2, ''Rod Sequence Control System;'' TS 3/4.10.2, ''Special Test Exceptions - Rod Sequence Control System;'' the Bases for TS 2.2.1.2, ''Average Power Range Monitor;'' the Bases for TS 3/4.1.4, ''Control Rod Program Controls;'' and the Bases for TS 3/4.10.2, ''Rod Sequence Control System.'' The changes eliminate the Rod Sequence Control System (RSCS) Limiting Condition for Operation and Surveillance Requirements from the TSs and reduce the Rod Worth Minimizer low power setpoint to 10% from 20%. Changes to other sections of the TSs delete reference to the RSCS from the TSs and incorporate additional requirements necessary to support the elimination of the RSCS.

Date of issuance: September 30, 1997

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

Amendment No.: 105

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

Date of initial notice in **Federal Register:** August 27, 1997 (62 FR 45462) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 30, 1997. No significant hazards consideration comments received: No.

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

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: May 28, 1997

Brief description of amendments: The amendments revise the Technical Specifications to clarify that testing of each shared emergency diesel generator (EDG), 1-2A and 1C, to comply with surveillance requirement 4.8.1.1.2.e is only required once per 5 years on a per EDG basis, not on a per unit basis.

Date of issuance: October 1, 1997

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

Amendment Nos.: 129, 122

Facility Operating License Nos. NPF-2 and NPF-8: Amendments revise the Technical Specifications.

Date of initial notice in **Federal Register:** June 18, 1997 (62 FR 33135) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 1, 1997. No significant hazards consideration comments received: No.Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50-321, Edwin I. Hatch Nuclear Plant, Unit 1, Appling County, Georgia

Date of application for amendment: May 9, 1997, as supplemented September 19, 1997

Brief description of amendment: The amendment revises the minimum critical power ratio safety limits for a mixed core of GE9B/GE12/GE13 fuel for Cycle 18 operation.

Date of issuance: October 8, 1997

Effective date: Prior to the restart from the Hatch Unit 1 outage currently scheduled to begin October 1997.

Amendment No.: 209

Facility Operating License No. DPR-57: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 30, 1997 (62 FR 40857) The September 19, 1997, submittal 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 October 8, 1997. No significant hazards consideration comments received: No.Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

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

Date of application for amendments: May 9, 1997, as supplemented September 3, 1997

Brief description of amendments: The amendments revise the applicability requirements for the Rod Block Monitor (RBM) to require that the RBM be operable whenever reactor thermal power is greater than or equal to 29 percent of rated thermal power.

Date of issuance: October 8, 1997

Effective date: As of the date of issuance to be implemented prior to Unit 1 startup from the fall 1997 refueling outage for Unit 1; and implemented within 30 days from issuance for Unit 2.

Amendment Nos.: 210, 151

Facility Operating License Nos. DPR-57 and NPF-5: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 30, 1997 (62 FR 40857) The September 3, 1997, submittal provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 8, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

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

Date of application for amendments: August 21, 1996, as supplemented by letters dated March 17, March 27, April 3, and July 15, 1997 (TS 96-07)

Brief description of amendments: The amendments change the Technical Specifications (TS) by revising the as-found setpoint tolerance band for the pressurizer Code safety relief valves and the main steam Code safety relief valves from plus or minus one percent to plus or minus three percent.

Date of issuance: September 29, 1997

Effective date: September 29, 1997

Amendment Nos.: 229 (Unit 1), 220 (Unit 2)

Facility Operating License Nos. DPR-77 and DPR-79: Amendments revise TS.

Date of initial notice in **Federal Register:** October 9, 1996 (61 FR 52969) The March 17, March 27, April 3, and July 15, 1997, letters provided clarifying information that did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 29, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: August 14, 1997 (TSCR 199)

Brief description of amendments: The amendments revise TS 15.4.2.B. ''In-Service Inspection and Testing of Safety Class Components Other than Steam Generator Tubes,'' to modify item 2 by deleting the reference to TS 15.4.4 and referencing the Containment Leakage Rate Testing Program; TS 15.6.12.A.1, ''Containment Leakage Rate Testing Program,'' to eliminate the one-time requirement for Unit 2 Type A testing since the testing has been completed; and TS Bases 15.4.4 to delete the specific bases for containment purge valve testing and to delete a reference that is no longer used. Date of issuance: September 29, 1997Effective date: September 29, 1997, with full implementation within 45 days

Amendment Nos.: 181 and 185

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

Date of initial notice in **Federal Register:** August 27, 1997 (62 FR 45466) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 29, 1997. No significant hazards consideration comments received: No.

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

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

Date of amendment request: July 29, 1997

Brief description of amendment: The amendment changes the wording of Action Statement 5a to Technical Specification Table 3.3-1, ''Reactor Trip System Instrumentation.'' This action statement prescribes a set of actions to be accomplished when a source range neutron detector is inoperable with the plant shutdown. The proposed wording change will clarify the times and order in which these actions are to be performed.

Date of issuance: September 29, 1997

Effective date: September 29, 1997, to be implemented within 30 days from the date of issuance.

Amendment No.: 111

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

Date of initial notice in **Federal Register:** August 27, 1997 (62 FR 45467) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 29, 1997. 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

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

Date of amendment request: September 6, 1997

Brief description of amendment: This amendment allows the testing of certain contacts in the emergency diesel generator load sequencer to be done with the unit at power (Mode 1) and provides an additional 24 hours to the time allowed by TS 4.0.3 to complete the testing.

Date of issuance: October 7, 1997

Effective date: October 7, 1997

Amendment No.: 112

Facility Operating License No. NPF-42. The amendment revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: Yes (62 FR 49261 dated September 19, 1997). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by October 20, 1997, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated October 7, 1997.

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037

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, Maryland, this 15th day of October 1997.

For the Nuclear Regulatory Commission

Elinor G. Adensam,

Acting DirectorDivision of Reactor Projects - III/IV, Office of Nuclear Reactor Regulation

[Doc. 97-27877 Filed 10–21–97; 8:45 am]

BILLING CODE 7590-01-F

NUCLEAR REGULATORY COMMISSION

[NUREG–1569]

Draft Standard Review Plan For In Situ Uranium Extraction License Applications

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability; opportunity for comment.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is soliciting comments on a Draft Standard Review Plan for *in Situ* Uranium Extraction License Applications (NUREG–1569) from interested parties. A NRC source and byproduct material license is required under the provisions of Title 10 of the Code of Federal Regulations, Part 40 (10 CFR Part 40), Domestic Licensing of Source Material, to recover uranium by *in situ* leach uranium extraction mining techniques (*in situ* leaching). An applicant for a new operating license, or for the renewal or amendment of an existing license, is required to provide detailed information on the facilities, equipment, and procedures to be used, and if appropriate, an environmental report that discusses the effect of proposed operations on public health and safety and on the environment. This information is used by Nuclear Regulatory Commission staff to determine whether the proposed activities will be protective of public health and safety and be environmentally acceptable. The purpose of this standard review plan is to provide NRC staff with specific guidance on the review of this information and will be used to ensure a consistent quality and uniformity of staff reviews. Each section in the review plan provides guidance on what is to be reviewed, the basis for the review, how the staff review is to be accomplished, what the staff will find acceptable in a demonstration of compliance with the regulations, and the conclusions that are sought regarding the applicable sections in 10 CFR. The review plan is also intended to improve the understanding of the staff review process by interested members of the public and the uranium recovery industry. The draft was developed using input from (1) staff review precedents; (2) staff inspection experiences; (3) public meetings with industry; and (4) experience from the State of Texas, which is an agreement state for uranium recovery and has 15 licensed *in situ* leach operations.

Opportunity to Comment: Interested parties are invited to comment on the review plan. Interested parties are also asked to comment on the level and extent that staff could rely on technical reviews performed by non-agreement states in areas where the NRC and the State have concurrent regulatory authority. These areas include land application, nonradiological soil cleanup, upper control limit, and groundwater restoration reviews. A final review plan will be prepared after the NRC staff has evaluated public comments received on the draft review plan.

DATES: Written comments must be received prior to December 8, 1997.

ADDRESSES: Comments on the draft review plan should be sent to the Chief, Rules and Directives, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

AVAILABILITY: A copy of the Draft Standard Review Plan (NUREG–1569) may be obtained by writing to the Printing and Graphics Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555–0001.

Dated at Rockville, Maryland, this 14th day of October 1997.

For the Nuclear Regulatory Commission.

Joseph J. Holonich,

Chief, Uranium Recovery Projects Branch, Division of Waste Management, Office of Nuclear Material, Safety and Safeguards.

[FR Doc. 97–28002 Filed 10–21–97; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Applications, Hearings, Determinations, Etc. Tivoli Industries, Inc.

October 16, 1997.

Issuer Delisting; Notice of Application to Withdraw from Listing and