Engineering Company, P.O. Box 101, Florham Park, NJ 07932.

On August 17, 1993, participants in the Petroleum Environmental Forum Project No. 93–02 filed their original notification pursuant to Section 6(a) of the Act. The Department of Justice published a notice in the **Federal Register** pursuant to Section 6(b) of the Act on September 23, 1993, 58 FR 49530. Additionally, a correction notice was published in the **Federal Register** on January 14, 1994, 59 FR 2439.

Constance K. Robinson,

Director of Operations, Antitrust Division.

[FR Doc. 95–2488 Filed 1–31–95; 8:45 am]

BILLING CODE 4410–01–M

DEPARTMENT OF LABOR

Office of the Secretary

All Items Consumer Price Index for all Urban Consumers United States City Average

Pursuant to Section 604(c) of the Motor Vehicle Information and Cost Savings Act, which was added to the Motor Vehicle Theft Law Enforcement Act of 1984, and the delegation of the Secretary of Transportation's responsibilities under that Act to the Administrator of the Federal Highway Administration (49 C.F.R., Section 501.2(f)), the Secretary of Labor has certified to the Administrator and published this notice in the Federal Register that the United States City Average All Items Consumer Price Index for All Urban Consumers (1967=100) increased 42.7 percent from its 1984 base period annual average of 311.1 to its 1994 annual average of 444.0.

Signed at Washington, D.C., on the 25th day of January 1995.

Robert B. Reich,

Secretary of Labor.

[FR Doc. 95–2453 Filed 1–31–95; 8:45 am]

BILLING CODE 4510–24–M

NATIONAL SCIENCE FOUNDATION

Conservation Act of 1978; Notice of Permit Modification

AGENCY: National Science Foundation.

SUMMARY: The Foundation modified a permit to conduct activities regulated under the Antarctic Conservation Act of 1978 (Public Law 95–541; Code of Federal Regulations Title 45, Part 670).

FOR FURTHER INFORMATION CONTACT: Peter Karasik, Permit Office, Office of Polar Programs, Rm. 755, National Science Foundation, 4201 Wilson Boulevard, Arlington, VA 22230.

DESCRIPTION OF PERMIT AND MODIFICATION: On September 7, 1994, the National Science Foundation issued a permit to Dr. Wayne Z. Trivelpiece after posting a notice in the August 8, 1994 **Federal Register**. Public comments were not received. A request to modify the permit was posed in the **Federal Register** on December 21, 1994. No public comments were received. The modification, issued by the Foundation on January 23, 1995, allows for the collection of 1 ml blood samples from 20 Adelie penguins breeding at Copacabana Station on King George Island and from 20 Adelie penguins breeding at Palmer Station on Anvers Island. All birds will be released after capture and collection of the blood samples.

LOCATION: SSSI#8—Western Shore Admiralty Bay, King George Island and Palmer Station vicinity, Anvers Island.

DATES: January 23, 1995—April 15, 1995.

Guy G. Guthridge,

Permit Office.

[FR Doc. 95–2474 Filed 1–31–95; 8:45 am]

BILLING CODE 7555–01–M

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 January 5, 1995, through January 20, 1995. The last biweekly notice was published on January 18, 1995 (60 FR 3669).

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 Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, 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 March 3, 1995, 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, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1–(800) 248–5100 (in Missouri 1–(800) 342–6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to *(Project Director)*: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, 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.

Boston Edison Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: November 22, 1994.

Description of amendment request: The proposed amendment would increase the current Emergency Diesel Generator (EDG) allowed out-of-service time in Specification 3.5.F from 72 hours to 7 days, deletes the daily testing of the operable diesel generator in Specification 4.5.F.1, when it is determined that the other diesel generator is inoperable, and revises specification 3.9.B.1 and 2 for EDG operability.

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

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

Operation of PNPS [Pilgrim Nuclear Power Station] in accordance with the proposed license amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Implementation of the proposed change is expected to result in an increase in the probability of core damage, from 5.85E–5/year (this is the PNPS IPE [individual plant examination] core damage frequency) to 5.88E–5/year. This increase is less than one percent and is considered to be insignificant relative to the underlying uncertainties involved with probabilistic risk assessments.

Deleting the testing requirement for an EDG when the other EDG is in repair does not increase the probability or consequences of an accident previously evaluated because the reliability program and Technical Specification required surveillances continue to provide the added assurance sought by the testing. The elimination of this testing might improve the overall reliability of the EDGs.

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

Operation of PNPS in accordance with the proposed license amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. No change is being made in the manner in which the EDG's provide plant protection. No new modes of plant operation are involved. Extending the EDG OOS [out of service] and, deleting the testing requirement for one EDG when the other EDG is in repair does not necessitate physical alteration of the plant or changes in plant operational limits.

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

Operation of PNPS in accordance with the proposed license amendment will not involve a significant reduction in a margin of safety. [***], incorporation of the proposed change involves an insignificant reduction in the margin of safety.

As previously stated, implementation of the proposed changes is expected to result in an insignificant increase in: (1) power unavailability to the emergency buses (given that a loss of offsite power has occurred), and (2) core damage frequency. EDG reliability improvement is expected due to increased quality and thoroughness of EDG maintenance. Implementation of the proposed changes does not increase the consequences of a previously analyzed accident nor significantly reduce a margin of safety. Functioning of the EDGs and the manner in which limiting condition of operability are established are unaffected.

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: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

Attorney for licensee: W.S. Stowe, Esquire, Boston Edison Company, 800 Boylston Street, 36th Floor, Boston, Massachusetts 02199.

NRC Project Director: Walter R. Butler.

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

Date of amendment request: December 27, 1994.

Description of amendment request: The requested Technical Specifications (TS) change relocates the turbine rotor inspection requirement, TS 4.1–3, Item 13, to the Updated Final Safety Analysis Report (UFSAR), Section 10.2. This TS requires a turbine inspection, including visual, magnaflux, and dye petrant inspections on a frequency of every five years with a maximum time between tests of six years.

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 requested change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The requested change relocates the turbine inspection requirement from the TS to the UFSAR. Turbine inspections will continue to be controlled and performed such that the low turbine missile generation probability will be maintained. The consequences of missile generation are unchanged since this change does not involve the addition or modification of plant equipment, nor does it alter the design or operation of plant systems. Therefore, there would be no increase in the probability or consequences of an accident previously evaluated.

2. The requested change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The requested change relocates the turbine inspection requirement from the TS to the UFSAR. Turbine inspections will continue to be controlled and performed such that the low turbine missile generation probability will be maintained. This change does not involve the addition or modification of plant equipment, nor does it alter the design or operation of plant systems. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The requested change does not involve a significant reduction in the margin of safety. The requested change relocates the turbine inspection requirement from the TS to the UFSAR. Turbine inspections will continue to be controlled and performed such that the low turbine missile generation probability will be maintained. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: R. E. Jones, General Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Project Director: William H. Bateman.

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

Date of amendment request: December 19, 1994.

Description of amendment request: The proposed one-time schedular extension would allow the third test of the first 10-year service period to be performed during refueling outage no. 7, at approximately a 54 month interval instead of the current maximum Technical Specification interval of 50 months, and coincident with the 10-year service period to be performed during refueling outage no. 7 and the 10-year inservice inspection,

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

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

This [extension] request applies to the ILRT [integrated leak rate testing] and does not affect the local leak rate testing of containment penetrations and isolation valves where the majority of the leakage occurs. The allowable containment leakage used in the accident analysis for offsite doses, L_a, is 0.1 [weight percent per day] and for conservatism the leakage is limited to 75 percent L_a at startup to account for the possible degradation of containment leakage barriers between two ILRT tests. Based on the "as left" leakage data for the past two ILRTs, the additional time period added to the testing interval would not adversely impact the containment leakage barriers to the extent that degradation would cause leakage to exceed that assumed in the accident analysis.

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

The change to the Surveillance Requirement is a one time [extension] to extend the surveillance interval from the maximum of 50 months to approximately 54 months for performance of the third ILRT in the first service period. There are no design changes being made that would create a new type of accident or malfunction and the method and manner of plant operation remain unchanged. Extension of the surveillance interval for performing the ILRT does not adversely impact the surveillances ability to show that containment integrity is maintained.

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

There are no changes being made to the safety limits or safety system settings that would adversely impact plant safety. The change is a one time [extension] to extend the time interval for performing an ILRT approximately four months beyond the current maximum interval. In addition to the indication of continued containment integrity provided by the Local Leak Rate Testing program, the surveillance test data from the first and second ILRTs illustrates that there is sufficient leakage margin to remain well below the allowable leakage rate of L_a. The as-left leakage rate for the last ILRT was 0.0614 [weight percent per day], which is well below the 0.075 [weight percent per day] allowed by the T.S., and therefore provides margin for degradation that is greater than the minimum provided by the Technical Specifications. Therefore, this change does not significantly reduce the margin of safety for Technical Specification 3.6.1.2.

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: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

Attorney for licensee: R. E. Jones, General Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Project Director: William H. Bateman.

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

Date of amendment request: December 29, 1994.

Description of amendment request: The proposed amendment would affect the method of controlling the pH of the post-LOCA containment sump solution by allowing the replacement of the existing operator actuated Iodine Removal System with a passive system of baskets of Trisodium Phosphate (TSP) in the lower regions of the containment. The current Iodine Removal System provides sodium hydroxide (NaOH) for injection into the containment spray to maintain pH of the sump solution.

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 following evaluation supports the finding that operation of the facility in accordance with the proposed change from NaOH to TSP requirements would not:

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

The substitution of TSP baskets for the NaOH addition equipment would not cause any changes to the capability, settings, or operation of the plant systems (other than the Iodine Removal System itself) and would not, therefore, have any effect on the probability of occurrence of an accident.

The substitution of TSP baskets for the NaOH addition equipment has the effect of providing more immediate control of post-LOCA sump pH, thereby increasing the assurance that iodine will remain in solution throughout a postulated event. The consequences of accidents evaluated in the FSAR [Final Safety Analysis Report] will not be increased by this increased assurance.

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

The TSP baskets are passive components which have no interaction with plant equipment unless flooding occurs in the containment. They are designed and located such that they will not interact with any plant safety equipment during a seismic event. The NaOH equipment, which will be replaced by the TSP baskets, has no function or effect on other equipment except during accident conditions. Therefore, the substitution of TSP baskets for NaOH addition equipment cannot create the possibility of a new or different kind of accident from any previously evaluated.

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

The substitution of TSP baskets for the NaOH addition equipment would assure that the sump pH at the initiation of RAS [recirculation actuation signal] is between 7.0 and 8.0 as assumed in the MHA [maximum hypothetical accident] analysis. Therefore, this change would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Van Wylen Library, Hope College, Holland, Michigan 49423.

Attorney for licensee: Judd L. Bacon, Esquire, Consumers Power Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Project Director: John N. Hannon.

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

Date of amendment request: December 7, 1994.

Description of amendment request: The amendments revise the Technical Specification action statement to allow the Control Room Air Intake to remain open when radiation monitors (EMF–43A and EMF–43B) are inoperable. Immediate action to return the monitors to service would be required.

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

1. The proposed amendment would not involve a significant increase in the probability or consequences of any accident previously evaluated in the FSAR [Final Safety Analysis Report].

The amendment change will ensure correct Control Room Ventilation system alignment in order to mitigate the consequence of a Design Basis LOCA as described in FSAR Section 15.6.5.3, Environmental Consequences of a Loss of-Coolant Accidents, Control Room Operator Dose.

The amendment change will permit the intake to remain open and will specify that action to repair the affected monitor shall be taken immediately. The change itself is not considered to be an initiator of any previously evaluated accident. Maintaining the VC intake open with an inoperable monitor will not result in any accidents that have not been previously evaluated. The implementation of immediate actions to repair the inoperable monitor does not in itself represent any accidents that have not been previously evaluated. Therefore, the proposed Technical Specification change does not increase the occurrence probability of previously evaluated accidents.

The change to permit maintenance of open intakes will not increase the consequences of any previously evaluated accidents. The proposed amendment change is consistent with the original Safety Analysis concerning the Dose to the Operators.

The analysis determined that the Doses to the Operators were within acceptable ranges given the assumptions that the intakes would remain open and the contaminated air was processed through a Safety Related filter train prior to introduction into the Control Room. The proposed change remains consistent with this analysis and does not change the assumptions or methodology utilized to assess the Doses to the Operators for a hypothesized DBA; therefore, the proposed amendment change will not increase the consequences of any previously evaluated accident.

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

The proposed change will not modify, delete, or add any systems or components; therefore, no new failure modes or accidents scenarios will be created.

No test or experiments will be revised; therefore, no new initiating events or unanalyzed condition will be created. Administrative changes to surveillance procedures will be minor and will not create a safety concern.

3. No significant reduction in a margin of safety will occur.

The proposed amendment change requiring immediate action to initiate repairs to an inoperable monitor does not impact existing Safety Margins. Since requirements for immediate corrective action does not currently exist within the Specification, the changes will enhance the availability of the subject monitors.

The proposed amendment does not change/impact any assumption or methods utilized to assess the doses to the operators for a hypothetical worst case DBA. Accordingly, the proposed amendment does not reduce any safety margins.

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: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242.

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: December 9, 1994.

Description of amendment request: The proposed amendment would revise the technical specifications (TSs) by revising the allowable opening tolerances on the Pressurizer Code Safety Valves and the Main Steam Line Code Safety Valves from plus or minus 1% to plus or minus 3%. This request is submitted as a result of an effort to improve valve performance and to ensure that the TS limits are consistent with expected valve performance capabilities.

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 any change to the physical characteristics of the PSVs [pressurizer safety valves] and MSSVs [main steam safety valves] and will have no impact on the PSVs and MSSVs as-left setting. This change only allows for a larger (plus or minus 3% versus plus or minus 1%) as-found setpoint tolerance. Therefore, this change has no impact on the probability of occurrence of any accident previously evaluated. The impact of this change on the FSAR [final safety analyses report] analyses has been evaluated and the results of the impacted events have been found to be within the acceptable limits.

Therefore, revising the PSV and MSSV as-found opening setpoint tolerance from plus or minus 1% to plus or minus 3% does not increase the probability or consequences of an accident previously evaluated.

2. The proposed changes to the PSVs and MSSVs as-found opening setpoint tolerance do not modify equipment or change the manner in which the plant will be operated. The safety valves will continue to function per their design. Since no hardware modifications or changes in operation procedures will be made, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The impact of the proposed changes on the Waterford 3 FSAR analyses have been evaluated. The evaluation demonstrates that the results of the impacted events remained within the acceptable limits. The system capabilities to mitigate and/or prevent accidents will be the same as they were prior to these changes. Therefore, the proposed changes do not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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, Louisiana 70122.

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

NRC Project Director: William D. Beckner.

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

Date of amendment request: December 9, 1994.

Description of amendment request: The proposed amendment would revise the technical specifications (TSs) by revising a plant protection system (PPS) trip setpoint and several allowable values such that they will be consistent with the current setpoint/uncertainty methodology being implemented at Waterford 3.

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

Implementing the proposed change will not affect any design basis accident. The revised Trip Setpoint and Allowable Values are based upon the same Analytical Limits that form the basis for the current Trip Setpoints and Allowable Values. The design basis for each Trip Setpoint was verified to be consistent with the appropriate accident analyses as part of the process of revising the PPS setpoint analysis. The proposed change would implement a new Trip Setpoint for the Reactor Coolant (RC) System Low Flow Reactor trip and new Allowable Values for RC Low Flow, HI Log Power, HI Steam Generator Water Level, HI Containment Pressure, Low Pressurizer Pressure, Low Steam Generator Pressure, Low Steam Generator Water Level, and Low RWSP [refueling water storage pool] Level, based on the results of calculation EC-I92–019. The revised Low RC Flow Trip Setpoint is based on the same analytical limit as the current setpoint. The revised calculation uses the same design inputs with a similarly based methodology to calculate a smaller loop uncertainty. This results in a revised RC Low Flow Trip Setpoint that retains the original analysis limit. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any previously analyzed accident.

Plant operation and the manner in which the plant is operated will not be altered as a result of implementing the proposed change since no new system or design change is being implemented. The proposed Setpoint and Allowable Value changes do not create any new system interactions or interfaces. All information used to calculate the new Trip Setpoint is consistent with that of the existing accident analyses, and no new system interfaces/interactions are created. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed setpoint change revised the point at which the RCS Low Flow reactor trip initiates a reactor trip. The Trip Setpoint is based on the same Analytical Limit used to determine the current setpoint. In addition, the same basic setpoint determination methodology is employed. That is, the Trip Setpoint is the Analytical Limit plus or minus the Total Loop Uncertainty [TLU]. The Allowable Value is the Trip Setpoint plus or minus the Periodic Test Error [PTE]. The change in the setpoint and allowable values are [sic] due to a change in calculated TLU and PTE. The proposed Trip Setpoint and Allowable Values are based on the same Analytical Limits for the affected parameters and are determined using approved methodology. Therefore, the proposed change will not involve a significant reduction in margin of safety.

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

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

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

NRC Project Director: William D. Beckner.

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: December 27, 1994.

Description of amendment request: The proposed amendments would revise the period for conducting leak testing of containment purge valves to every 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:

1. The proposed change to the Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously evaluated because the [test] results have demonstrated that the resilient seat material does not degrade and cause containment isolation valves to leak. Therefore the valves will perform as assumed in the accident analyses.

2. The proposed change to the Technical Specifications does not create the possibility of a new or different kind of accident from any accident previously evaluated because it does not require the valves to function in any manner other than that which is currently required.

3. The proposed addition to the Technical Specifications does not involve a significant reduction in a margin of safety because it only affects the frequency of the test and does not change the leakage acceptance criteria. Since sufficient data has been collected to demonstrate that the resilient seals do not degrade, testing at the same frequency as other containment isolation valves will not reduce the margin of safety provided by the Technical Specifications.

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: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia 30830.

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308.

NRC Project Director: Herbert N. Berkow.

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: December 29, 1994.

Description of amendment request: This request withdraws a similar request dated January 22, 1993, as supplemented August 8, 1993, and submits a new one in its place. The proposed amendments would revise the Technical Specifications (TS) to add the automatic load sequencer specification to TS Section 3/4.3, Instrumentation, and associated Bases, and TS Section 3/4.8, Electrical Power Systems.

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 to the Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously evaluated because the action to be taken when an automatic load sequencer is inoperable is consistent with that of a more stringent condition already specified, namely, the loss of an entire train of emergency power during Modes 1–4, and for Modes 5 and 6 adding specific actions which previously had never been addressed in TS.

2. The proposed change to the Technical Specifications does not create the possibility of a new or different kind of accident from any accident previously evaluated because it does not involve any change to the design, operation, or performance of the automatic load sequencer. It only serves to clearly identify the appropriate conservative response to an inoperable automatic load sequencer applicable to the plant mode of operation.

3. The proposed change to the Technical Specifications does not involve a significant reduction in a margin of safety because the proposed actions to take when an automatic load sequencer is inoperable [are] the same as the action already required by the Technical Specifications when no power is available to the entire emergency bus during Modes 1–4 and by adding requirements during Modes 5 and 6, which had previously never been addressed.

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: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia 30830.

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308.

NRC Project Director: Herbert N. Berkow.

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: January 3, 1995.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) with editorial changes to the Action Statements of TS Sections 3.8.1.1 and 3.8.1.2 in order to reflect the availability of a third offsite ac electrical source. Surveillance Requirement 4.8.1.1.1 is being clarified to distinguish that the offsite ac circuits which are connected to the onsite Class 1E distribution system are required to be verified OPERABLE. The amendments also modify the Technical Specifications with the addition of a footnote to TS Section 3.8.3.1, to allow the connection of the third offsite ac source to the onsite busses.

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

Based on the considerations regarding the addition of a footnote for proper bus alignment during operating conditions, the licensee submitted the following analysis in accordance with 10 CFR 50.92.

1. The proposed change to the Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously evaluated because the probability of an LOSP or an SBO is not increased by the allowance of having both redundant emergency busses of 4160 volt switchgear connected to one offsite source (RAT). The probability of having an LOSP is not increased since the TS currently allow for a 72 hour LCO for one offsite power source and the time the two redundant 4160 volt safety busses will be temporarily aligned to one RAT is well within this time frame. During this time the busses are interconnected, each bus is provided adequate protection and separation by having separate and redundant Class 1E circuit breakers, one per bus. The probability of an SBO is not increased since neither bus' EDG will be affected during this operation, and since this is a proceduralized manual alignment, the interconnection to one RAT will not be initiated if either EDG were inoperable. Also, the addition of the new "swing" offsite power source (SAT), increases availability and flexibility of the VEGP response to either an LOSP or SBO.

2. The proposed change to the Technical Specifications does not create the possibility of a new or different kind of accident from any accident previously evaluated because the only postulated adverse consequences of tying both redundant 4160 volt safety busses together to one RAT is an LOSP. An LOSP is a design basis event which has already been analyzed for VEGP. In response to an LOSP, both EDGs remain capable of carrying the required loads to mitigate the consequences of any postulated design basis accident during or coincident with an LOSP.

3. The proposed addition to the Technical Specifications does not involve a significant reduction in a margin of safety because the only accident mitigating equipment and/or power sources which will be unavailable during the transfer of offsite power sources is the offsite power source being removed from service, allowed by existing TS LCO 3.8.1.1(a). The 13.8 kV loads associated with the RAT being removed from service and all of the 4160 volt non-Class 1E loads fed from either RAT will be unavailable during this temporary alignment. All of these loads are nonsafety related and therefore are enveloped by the existing LOSP analysis.

Based on the considerations regarding clarification of SAT Use and Expanded Bases, the licensee submitted the following analysis in accordance with 10 CFR 50.92.

1. The proposed change to the TS does not involve a significant increase in the probability or consequences of an accident previously evaluated because only clarifications to existing TS action statements and an additional expanded bases are being made. No changes to the existing TS requirements for A.C. sources are being made. The safety function of the offsite power source is unchanged by the addition of the SAT and the probability of an LOSP or SBO is not increased. In actuality, the addition of the SAT increases the availability and flexibility of VEGP responses to either an LOSP or SBO.

2. The proposed change to the TS does not create the possibility of a new or different kind of accident from any accident previously evaluated because the loss of the SAT while being utilized to meet TS offsite power source requirements is enveloped by existing LOSP analysis.

3. The proposed change does not involve a significant reduction in a margin of safety because although the SAT has no 13.8 kV secondary winding, nor the same capacity as a RAT for accepting 4.16 kV non Class 1E loads, these loads are nonsafety related and therefore enveloped by existing analysis. If a unit trip were to occur while one 4.16 kV safety bus is being powered from the SAT, the effect is a loss of the 13.8 kV and non Class 1E 4.16 kV loads associated with the out of service RAT. This scenario is enveloped by existing LOSP analysis.

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: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia 30830.

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308

NRC Project Director: Herbert N. Berkow.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan

Date of amendment requests: August 12, 1992 and supplemented April 12, 1993.

Description of amendment requests: The proposed amendments would change the minimum channels operable for the pressurizer safety valve position indicator acoustic monitor to two out of three total from one per valve. The amendments also delete footnotes which are no longer applicable.

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

We [the licensee] have evaluated the proposed T/Ss exemption and have determined that it should not require a significant hazards consideration based on the criteria established in 10CFR50.92(c). Operation of the Cook Nuclear Plant in accordance with the proposed amendment will not:

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

Although the proposed exemption results in the operator having one less source of information on plant status, it does not create a significant increase in the probability or consequences of an accident previously evaluated. The acoustic monitors do not perform a function vital to safe shutdown or to the isolation of the reactor, or the reactor coolant system pressure boundary, nor is there a mechanism involving an operable or inoperable pressurizer safety valve acoustic monitor which would initiate an accident. These monitors were added to meet the requirements of NUREG–0578 and NUREG–0737. During normal operations, other instrumentation exists that provides the operator with indication of safety valve actuation. The acoustic monitors are not necessary to and are not used in the emergency operating procedures. In addition, the acoustic monitors being inoperable will not result in an uncontrolled release of radiation to the environment and will not initiate an accident. Finally, although the operator may have one less channel operable, the operator receives no less information than if all three channels are operable because one valve opening causes all operable channels to actuate. Therefore, we conclude that the proposed T/Ss changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

As previously stated, the purpose of the acoustic monitor is to provide the operator with information regarding safety valve position that may assist in the mitigation of the consequences of an accident. Specifically, it provides information that a safety valve has lifted. However, the operator has other mechanisms for obtaining equivalent information. In addition, the signals generated by an acoustic monitor do not initiate any other equipment actuation, nor will the inoperability of an acoustic monitor initiate any accident. Consequently, the proposed T/Ss changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed T/Ss changes result in the operator potentially having one less source of information on plant status. However, we believe the margin of safety is not reduced for several reasons. First, the operator is provided with other viable flow detection devices to determine pressurizer safety valve position, i.e., the temperature sensor on the discharge line associated with the inoperable acoustic monitor, and pressurizer relief tank level (NLA–351), temperature (NTA–351) and pressure (NPA–351) indications. Also, the acoustic monitors are not used by the operators in an emergency situation, as the operator relies on other indications of loss of reactor coolant inventory per the emergency operating procedures. In addition, previous experience with the pressurizer safety valve position indicator acoustic monitoring system has shown that, when any one of the pressurizer safety valves opens, all three safety valve position indicator acoustic monitors are actuated. Because of this, the operator receives no less information regardless if only two or three channels are operable.

Based on the above, we believe that having an acoustic monitor inoperable does not warrant reactor and plant shutdown. As the T/Ss are currently stated, should one pressurizer safety valve position indicator acoustic monitor become inoperable, it must be restored to operable status within thirty days or the unit must be in hot shutdown within the subsequent twelve hours. Thermal cycling from unwarranted plant shutdowns increases the likelihood of reactor vessel embrittlement and unnecessarily challenges the safety systems. Because a signal from the pressurizer safety valve position indicator acoustic monitors is not necessary nor used to ensure the safe shutdown of the unit even if a pressurizer safety valve is opened or stuck open during an emergency situation, we believe that a plant shutdown due to an inoperable acoustic monitor would be unwarranted.

We believe that the unit can be operated safely and that we would still meet the intent of NUREG–0538 and NUREG–0737 with only two out of three pressurizer safety valve position indicator acoustic monitors operable.

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: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085.

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

NRC Project Director: John N. Hannon.

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

Date of amendment request: November 18, 1994.

Description of amendment request: The proposed amendment would change the title of certain Plant Operation Review Committee (PORC) members to reflect recent Maine Yankee organizational changes; update training requirements to comply with 10 CFR 50.120, Training and qualification of nuclear power plant personnel; and reporting frequency requirements for the Radioactive Effluent Release and Estimated Dose and Meteorological Summary Reports.

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). A summary of the licensee's analysis is presented below:

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

The changes proposed by this amendment request are administrative in nature. Because the proposed changes do not involve any physical alterations to plant equipment, operating setpoints, parameters or conditions, the plant's response to previously evaluated accidents is not affected.

The licensee therefore concludes that implementation of the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The administrative nature of the proposed changes does not affect the design, operation, maintenance or testing of the plant. Thus, no new modes of failure are created.

The licensee therefore concludes that implementation of the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change reflects an organizational change that does not modify the qualification requirements or competence of the members of the PORC. Thus, the capability of PORC to meet its responsibilities in accordance with the plant Technical Specifications is unchanged.

Deleting the current training requirement for Shift Technical Advisors eliminates duplicative training requirements and represents conformance to 10 CFR 50.120, Training and qualification of nuclear power plant personnel.

Elevating the responsibility for training the plant staff from the Manager, Operations Department, to the Vice President of Operations, does not represent a reduction in a margin of safety.

The proposed change to the Radioactive Effluent Release and Estimated Dose and Meteorological Summary Reports is related to the submittal schedule for statistical data and is administrative in nature. The change in submittal frequency provides consistency between the various required reports and also is administrative in nature.

The licensee therefore concludes that implementation of the proposed change would not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, Maine 04578.

Attorney for licensee: Mary Ann Lynch, Esquire, Maine Yankee Atomic Power Company, 329 Bath Road, Brunswick, Maine 04011.

NRC Project Director: Walter R. Butler.

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

Date of amendment request: December 16, 1994.

Description of amendment request: The proposed change to the Technical Specifications would require the wind direction and wind speed sensors at the 142 foot elevation to identify the data to determine action required to preclude flood damage to the Service Water Pumps. Also, the proposed change would correct a typographical error in the location of the sensors at the 374 foot elevation.

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

* * * The proposed changes do not involve a significant hazards consideration because the changes would not:

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

NNECO [Northeast Nuclear Energy Company] is proposing to revise LCOs [Limiting Conditions for Operation] 3.7.5.1.b.3 and 3.7.5.1.b.4 and Table 3.3–8 of the Millstone Unit No. 2 Technical

Specifications by changing the elevation that the average wind speed and average wind direction are measured and by correcting a typographical error, respectively. The proposed changes have no effect on any of the accidents analyzed in Chapter 14 of the Millstone Unit No. 2 FSAR [Final Safety Analysis Report]. Site flooding is considered in Section 2.5.4.2.1 of the FSAR. Utilizing the wind speed indicator at the 142-foot elevation, in lieu of the indicator on the 374-foot elevation will not significantly change the ability of personnel to predict the potential for a major storm with flooding.

The proposed changes do not alter the intent of the surveillances, do not involve any physical changes to the plant, do not alter the way any structure, system, or component functions, and do not modify the manner in which the plant is operated.

Based on the above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

NNECO is proposing to revise LCOs 3.7.5.1.b.3 and 3.7.5.1.b.4 and Table 3.3–8 of the Millstone Unit No. 2 Technical Specifications by changing the elevation that the average wind speed and average wind direction are measured and by correcting a typographical error, respectively. The proposed changes do not alter the intent of the surveillances, do not involve any physical changes to the plant, do not alter the way any structure, system, or component functions, and do not modify the manner in which the plant is operated.

While the proposed changes to LCOs 3.7.5.1.b.3 and 3.7.5.1.b.4 do change the measurement location stipulated by the technical specifications, this change is insignificant. Utilizing the wind speed indicator at the 142-foot elevation, in lieu of the indicator on the 374-foot elevation will not significantly change the ability of personnel to predict the potential for a major storm with flooding.

Based on the above, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

NNECO is proposing to revise LCOs 3.7.5.1.b.3 and 3.7.5.1.b.4 and Table 3.3–8 of the Millstone Unit No. 2 Technical Specifications by changing the elevation that the average wind speed and average wind direction are measured and by correcting a typographical error, respectively. The proposed changes will have no impact on the physical protective boundaries (fuel matrix/cladding, reactor coolant system pressure boundary, and containment). The proposed changes do not alter the intent of the surveillances, do not involve any physical changes to the plant, do not alter the way any structure, system, or component functions, and do not modify the manner in which the plant is operated.

While the proposed changes to LCOs 3.7.5.1.b.3 and 3.7.5.1.b.4 do change the manner in which potential flooding is predicted, this change is insignificant. Utilizing the wind speed and direction indicators at the 142-foot elevation, in lieu of the indicators at the 374-foot elevation will not significantly change the ability of personnel to predict the potential for a major storm with flooding.

Based on the above, the proposed changes do not involve a significant reduction in a margin of safety.

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

Local public document room location: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141–0270.

NRC Project Director: Phillip F. McKee.

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

Date of amendment request: December 21, 1994.

Description of amendment request: Proposed revision to License Condition and Technical Specifications to relocate the Fire Protection Requirements from the Technical Specifications to another controlled document, the technical requirements manual (TRM).

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

* * * The proposed changes do not involve a significant hazards consideration because the changes would not:

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

The proposed changes relocates the provisions of the Fire Protection Program that are contained in the Technical Specifications and places them in the TRM. No current requirements are being added or deleted aside from removal of the special reports section. Review of the Fire Protection Program and its revisions will be the responsibility of the PORC [Plant Operations Review Committee] and SORC [Station Operations Review Committee], just as it has always been the responsibility of these groups to review changes to the fire protection Limiting Condition for Operation and Surveillance Requirements when they were part of the Technical Specifications. In addition, no design basis accidents are affected by this change, nor are safety systems adversely affected by the changes. Therefore, there is no impact on the probability of occurrence or the consequences of any design basis accidents.

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

The proposed changes relocate the provisions of the Fire Protection Program that are contained in the Technical Specifications and places them in the TRM. No current requirements are being added or deleted aside from removal of the special report section. There are no new failure modes associated with the proposed changes. Since the plant will continue to operate as designed, the proposed changes will not modify the plant response to the point where it can be considered a new accident.

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

No change is being proposed for the Fire Protection Program requirements themselves. The relevant Technical Specifications are being relocated, and the requirements contained therein are being incorporated into the TRM. Plant procedures will continue to provide the specific instructions necessary for the implementation of the requirements, just as when the requirements resided in the Technical Specifications. Fire Protection Program changes will be governed by the provisions of 10 CFR 50.59 and the current fire protection license condition. As such, the changes do not directly affect any protective boundaries nor does it impact the safety limits for the boundary. Thus, there are no adverse impacts on the protective boundaries, safety limits, or 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: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141–0270.

NRC Project Director: Phillip F. McKee.

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

Date of amendment request: December 2, 1994.

Description of amendment request: The proposed amendment modifies the surveillance requirements for the power range neutron flux instrumentation to permit entering reactor operating modes 1 and 2 to perform necessary test for power range detectors.

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 (SHC), which is presented below:

* * * The proposed changes do not involve an SHC because the changes would not:

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

NNECO is proposing to modify Table 4.3–1 by adding Note 5 to Functional Units 2b, 3, and 4. This note provides an exception from the provisions of Technical Specification 4.0.4. Entry into Mode 2 or Mode 1, as appropriate, would allow for appropriate test conditions to complete the channel calibration of power range neutron detectors (i.e., Functional Units 2b, 3, and 4 of Table 4.3–1). This will improve plant safety by performing tests at proper conditions. The acceptance criteria, such as response times, test frequency, or test methods, are not revised. Therefore, the power range neutron detectors will perform their intended function when called upon. Additionally, the proposed changes are consistent with the new, improved STS for the Westinghouse plants (NUREG–1431).

Based on the above, the proposed changes to Functional Units 2b, 3, and 4 of Table 4.3–1 of the Millstone Unit No. 3 Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

The proposed changes to Functional Units 2b, 3, and 4 of Table 4.3–1 do not make any physical or operational changes to existing plant structures, systems, or components. The proposed changes do not introduce any new failure mode. They simply allow tests to be performed at appropriate conditions (e.g., Mode 2 or Mode 1 rather than Mode 4 or Mode 3).

Additionally, the proposed changes do not modify the acceptance criteria for the tests. The purpose of the tests is to ensure that the power range neutron detectors can perform their intended function.

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

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

The proposed changes to Functional Units 2b, 3, and 4 of Table 4.3–1 do not have any adverse impact on the design basis accident analyses. The applicable acceptance criteria for the power range neutron detectors will not be modified by the proposed changes. The proposed changes will permit the tests to be conducted under the proper conditions, so that the ability of the power range neutron detectors to perform their intended safety function can be confirmed.

Based on the above, there is no 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: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141–0270.

NRC Project Director: Phillip F. McKee.

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: August 30, 1994.

Description of amendment requests: The proposed amendments would revise the Technical Specifications (TS) for Prairie Island Nuclear Generating Plant as recommended by Generic Letter (GL) 93–05, ''Line-Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation.'' The proposed amendments would also revise testing and calibration requirements associated with the containment hydrogen recombiners. The proposed TS changes are as follows:

(1) TS Table 4.1–1C, ''Miscellaneous Instrumentation Surveillance Requirements.'' Delete Item 14, ''Accumulator Level and Pressure'' and corresponding frequency interval designations.

(2) TS Table 4.1–2A, ''Minimum Frequencies For Equipment Tests,'' Item 2. Revise the frequency for partial movement of all control rod assemblies from every 2 weeks to once per quarter.

(3) TS 4.3, ''Primary Coolant System Pressure Isolation Values.'' Under Specification heading, extend the amount of time the plant can be shut down before pressure isolation valve testing will be required from 72 hours to 7 days.

(4) TS SR 4.4.I, 4.4.I.a, 4.4.I.b, 4.4.I.b.1, 4.4.I.b.2, and 4.4.I.b.3, ''Electrical Hydrogen Recombiners.'' Revise the containment hydrogen recombiner testing surveillance frequency from every 6 months to every refueling interval. Delete the specific requirement to perform CHANNEL CALIBRATION of recombiner instruments and control circuits. Delete the requirement to sequentially perform the resistance to ground test following the functional test.

(5) TS SR 4.5.A.2.b, ''Containment Spray System.'' Revise the containment spray system nozzle testing surveillance frequency from once every 5 years to once every 10 years.

(6) TS SR 4.8.A.1, 4.8.A.2, and Footnote, ''Auxiliary Feedwater System.'' Revise the testing frequency for the auxiliary feedwater pumps from intervals of 1 month to semi-quarterly on a staggered test basis.

(7) BASES 4.8, ''Steam And Power Conversion Systems.'' Revise the Bases to include testing frequency for the auxiliary feedwater pumps from intervals of 1 month to semi-quarterly on a staggered test basis.

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.

Except for hydrogen recombiner changes to conform to Standard Technical Specifications, the requested changes were extensively reviewed by the NRC during the preparation of NUREG–1366 and Generic Letter 93–05. For the sake of clarity each proposed change is discussed separately in the order appearing in the Prairie Island Technical Specifications.

A. This Technical Specification amendment removes the accumulator water level and pressure channel surveillance from the Technical Specifications and places them into a licensee controlled test procedure. These changes are consistent with industry recognition that accumulator instrumentation operability is not directly related to the capability of the accumulators to perform their safety function.

Relocating the instrumentation surveillance requirements is an administrative change which will not affect equipment testing, availability, or operation. Therefore, it will not have an effect on the probability or consequences of an accident.

B. This Technical Specification amendment changes control rod movement from every two weeks to once every quarter. Control rod movement testing is performed to determine if the control rods are immovable. Control rods may be electrically stuck due to a problem in the control rod drive circuitry or mechanically stuck. Electrical problems with the control rod drive system, in general, do not prevent insertion of a control rod into the core when the reactor trip breakers are opened.

NUREG–1366 determined that control rod movement testing is not effective in determining immovable control rods. Most of the mechanically immovable control rods are discovered during plant startup during initial pulling of the rods or during rod drop testing. Extending the surveillance interval will not affect this failure discovery method.

The accident analyses assume that the single highest worth rod is struck while fully withdrawn and will not insert. One immovable control rod will still bound this accident analysis. For these reasons, the extension of the surveillance frequency from once every two weeks to once every quarter will not involve a significant increase in the probability or consequences of a previously evaluated accident.

C. This Technical Specification amendment will require Reactor Coolant Systems Pressure Isolation Valves (PIV) to be surveillance tested after seven days at cold shutdown instead of after three days at cold shutdown.

The PIVs are important in preventing over pressurization and rupture of the Emergency Core Cooling System low pressure piping which could result in a LOCA [loss-of-coolant accident] that bypasses containment. Allowable leakage from any PIV is sufficiently low to ensure early detection of possible in-series check valve failure. This change will not change the refueling outage surveillance, nor will it change the required testing to be performed after maintenance, repair, or replacement. The proposed level of surveillance is appropriate for these valves.

These valves have had very good operating performance and should continue to have the same performance record with continuation of the same maintenance and testing program. Furthermore, these valves are backed by motor or air-operated valves which have performed reliably.

For these reasons, the extension of the amount of time from three days to seven days before pressure isolation valve testing is required will not result in a significant increase in the probability or consequences of a previously evaluated accident.

D. This Technical Specification amendment will revise the containment hydrogen recombiner testing surveillance from every six months to every refueling interval.

The two independent containment hydrogen recombiners provide post-accident hydrogen control of the containment atmosphere. The recombiners are designed to be passive until an accident occurs.

Industry experience and in particular, Prairie Island experience has demonstrated that this equipment is highly reliable. Since the recombiners are not required until after an accident, there would likely be time to effect accessible repairs if the equipment were not operable.

Relocation of the recombiner calibration is an administrative change which will not affect recombiner operability. Deletion of specific testing sequence will not affect the performance of recombiner testing.

Equipment redundancy, reliability and time for repairs ensures post-accident control. For these reasons, these changes will not result in a significant increase in the probability or consequences of a previously evaluated accident.

E. This Technical Specification amendment will revise the containment spray system nozzle testing surveillance from once every five years to once every ten years.

Two independent containment spray systems provide post-accident cooling of the containment atmosphere and provide a mechanism for removing iodine from the containment atmosphere. This surveillance test verifies by air flow test that the spray nozzles are unobstructed. The extension of the surveillance frequency does not affect administrative controls that preclude entry of foreign material into the nozzles.

At Prairie Island the piping headers and nozzles are fabricated from austenitic stainless steel. There have been no reported in-service problems noted with spray nozzle testing from plants with stainless steel headers and nozzles and there is no indication that the lines would corrode and become obstructed.

For these reasons, this change will not result in a significant increase in the probability or consequences of a previously evaluated accident.

F. This Technical Specification amendment will revise the frequency for testing the Auxiliary Feedwater Pumps (AFWP) from monthly to semi-quarterly on a STAGGERED TEST BASIS.

Two 100% redundant, diverse pumps provide an emergency source of feedwater to the steam generators. The Prairie Island AFWPs have performed reliably. However, frequent testing of the pumps and associated equipment wears out the equipment resulting in equipment unavailability. AFWP availability will be increased by semi-quarterly surveillance testing on a STAGGERED TEST BASIS.

For these reasons, this change will not result in a significant increase in the probability or consequences of previously evaluated accident.

Therefore, the probability or consequences of an accident previously evaluated are not affected by any of the proposed amendments.

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

The extension of facility surveillance intervals as discussed previously will not result in changes in plant configuration or operation. The changes in recombiner calibration and testing will not result in changes in plant configuration or operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated would not be created.

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

The amendments proposed in this License Amendment Request do not reduce the ability of any system or component to perform its safety related function. The basis of NUREG–1366, Generic Letter 93–05, and the analysis performed in support of this License Amendment Request is that the reduction in surveillance testing can improve safety by reducing challenges to plant systems, personnel exposure, and equipment wear or degradation. The proposed changes to surveillance frequencies do not change the method of performing any surveillance. The operation of systems and equipment remains unchanged. Therefore, a significant reduction in the margin of safety would not be 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 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.

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: December 5, 1994.

Description of amendment requests: The proposed amendments would revise Technical Specification 3.8 to allow containment airlock doors to remain open during core alterations provided certain conditions are met. This request is similar to the amendment for Calvert Cliffs Nuclear Power Plant which the NRC approved on August 30, 1994. In addition, these amendments would allow containment penetrations to remain open during core alterations provided certain conditions are met.

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

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

The proposed containment refueling integrity amendments do not affect the probability of a fuel handling accident, they only deal with the containment systems.

The containment is provided for the purpose of mitigating the consequences of postulated accidents. For the fuel handling accident in containment, the licensing basis analyses, including the NRC safety evaluation report transmitted February 2, 1982, assumed that containment was completely abrogated and all radioactive materials released from the containment refueling pool are assumed to be released to the outside atmosphere. The requested amendments to Technical Specification 3.8.A.1.a modify the use of containment to mitigate the consequences of a fuel handling accident in containment, however, since instantaneous offsite release of all fuel handling accident materials released to containment has already been considered, the probability and consequences of a loss of containment accident are not increased.

Therefore, the probability or consequences of an accident previously evaluated are not affected by any of the proposed amendments.

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

The requested amendments to Technical Specification 3.8.A.1.a modify the use of containment to mitigate the consequences of a fuel handling accident in containment. There are no new failure modes or mechanisms associated with the proposed changes, nor do the proposed changes involve any modification of plant equipment or changes in plant operational limits. Previous analyses, including the NRC fuel handling accident safety evaluation for Prairie Island, have already assumed the containment is abrogated. The proposed license amendments may affect the release path for fission products released during a fuel handling accident in containment, but no new or different kind of accident will result.

Therefore, the possibility of a new or different kind of accident from any accident previously evaluated would not be created.

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

The margin of safety as defined by the licensing bases fuel handling accident analyses is not reduced. The previous analyses are very conservative, assuming all radioactive material released from [containment] by the fuel handling accident is immediately released to the outside atmosphere, and bound any changes introduced by these requested amendments.

Technical Specification 3.8.A.1.a exists to minimize the consequences of a fuel handling accident in containment. However, with the current Technical Specification 3.8.A.1.a, there will still be releases due to the necessity to open the containment airlocks to evacuate personnel. With implementation of this amendment, the ability of the closed airlocks to contain the accident releases may improve.

Some radioactive material could be released through containment penetrations that are open at the time of the accident. Since it is not likely that containment will be pressurized by a fuel handling accident, the releases are expected to be minimal. This amendment will maintain containment post-fuel handling accident offsite releases well within the limits of 10CFR100 and the current license basis releases.

Therefore, a significant reduction in the margin of safety would not be 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 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.

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: January 9, 1995.

Description of amendment requests: The proposed amendments would revise Prairie Island Nuclear Generating Plant Technical Specification (TS) 4.12, ''Steam Generator Tube Surveillance,'' to incorporate revised acceptance criteria for steam generator tubes with degradation in the tubesheet roll expansion region. These criteria for steam generator tube acceptance were developed by Westinghouse Electric Corporation and are known as F* (''F-Star'') and L* (''L-Star''). These criteria would be utilized to avoid unnecessary plugging and sleeving of steam generator tubes.

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

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

The supporting technical and safety evaluations of the subject criterion demonstrate that the presence of the tubesheet will enhance the tube integrity in the region of the hardroll by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and tube collapse is strengthened by the presence of the tubesheet in that region. The results of hardrolling of the tube into the tubesheet is an interference fit between the tube and the tubesheet. Tube rupture cannot occur because the contact between the tube and tubesheet does not permit sufficient movement of tube material. The radial preload developed by the rolling process will secure a postulated separated tube end within the tubesheet during all plant conditions. In a similar manner, the tubesheet does not permit sufficient movement of tube material to permit buckling collapse of the tube during postulated LOCA loadings.

The F* length of roll expansion is sufficient to preclude tube pullout from tube degradation located below the F* distance, regardless of the extent of the tube degradation. The existing Technical Specification leakage rate requirements and accident analysis assumptions remain unchanged in the unlikely event that significant leakage from this region does occur. As noted above, tube rupture and pullout is not expected for tubes using the F* criterion. Any leakage out of the tube from within the tubesheet at any elevation in the tubesheet is fully bounded by the existing steam generator tube rupture analysis included in the Prairie Island Plant USAR [Updated Safety Analysis Report]. For plants with partial depth roll expansion like Prairie Island, a postulated tube separation within the tube near the top of the roll expansion (with subsequent limited tube axial displacement) would not be expected to result in coolant release rates equal to those assumed in the USAR for a steam generator tube rupture event due to the limited gap between the tube and tubesheet. The proposed plugging criterion does not adversely impact any other previously evaluated design basis accident.

Leakage testing of roll expanded tubes indicates that for roll lengths approximately equal to the F* distance, any postulated faulted condition primary to secondary leakage from F* tubes would be insignificant.

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

Implementation of the proposed F* criterion does not introduce any significant changes to the plant design basis. Use of the criterion does not provide a mechanism to initiate an accident outside of the region of the expanded portion of the tube. Any hypothetical accident as a result of any tube degradation in the expanded portion of the tube would be bounded by the existing tube rupture accident analysis. Tube bundle structural integrity will be maintained. Tube bundle leaktightness will be maintained such that any postulated accident leakage from F* tubes will be negligible with regards to offsite doses.

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

The use of the F* criterion has been demonstrated to maintain the integrity of the tube bundle commensurate with the requirements of Reg Guide 1.121 [''Bases for Plugging Degraded PWR Steam Generator Tubes''] (intended for indications in the free span of tubes) and the primary to secondary pressure boundary under normal and postulated accident conditions. Acceptable tube degradation for the F* criterion is any degradation indication in the tubesheet region, more than the F* distance below the bottom of the transition between the roll expansion and the unexpanded tube. The safety factors used in the verification of the strength of the degraded tube are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in steam generator design. The F* distance has been verified by testing to be greater than the length of roll expansion required to preclude both tube pullout and significant leakage during normal and postulated accident conditions. Resistance to tube pullout is based upon the primary to secondary pressure differential as it acts on the surface area of the tube, which includes the tube wall cross-section, in addition to the inner diameter based area of the tube. The leak testing acceptance criteria are based on the primary to secondary leakage limit in the Technical Specifications and the leakage assumptions used in the USAR accident analysis.

Implementation of the tubesheet plugging criterion will decrease the number of tubes which must be taken out of service with tube plugs or repaired with sleeves. Both plugs and sleeves reduce the RCS (reactor coolant system) flow margin; thus, implementation of the F* criterion will maintain the margin of flow that would otherwise be reduced in the event of increased plugging or sleeving.

Based on the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety as defined in the USAR or the Technical Specification Bases.

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.

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: January 13, 1995.

Description of amendment requests: The proposed amendments would revise Prairie Island Nuclear Generating Plant Technical Specification 4.4.D.1 to change the interval for the performance of the Residual Heat Removal (RHR) System leakage test from once every 12 months to perform the test during each refueling shutdown.

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

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

The proposed changes to the RHR system leakage test interval only involve the leak-tightness of the RHR system for postaccident operation. As such, the proposed changes will have no impact on the probability of an accident previously evaluated.

The extension of the RHR system leakage test interval could increase the possibility of undetected RHR system leakage outside the containment during post accident conditions. However, the possible consequences of leakage from the RHR system outside containment are minor relative to those of the design basis accident. Therefore, because leakage from the RHR system has a minor effect on offsite dose, and since previous testing on a 12 month interval has not found significant RHR system leakage, the extension of the test interval to refueling is not expected to significantly impact the offsite dose consequences of an accident. In addition, it is probable that RHR system leakage would be identified during the normal quarterly functional testing and inspection of the RHR system.

Therefore, for the reasons discussed above, the proposed changes will not significantly affect the probability or consequences of an accident previously evaluated.

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

There are no new failure modes or mechanisms associated with the proposed changes. The proposed changes do not involve any modification of the plant equipment or any changes in operational limits. The proposed changes only modify the interval for the performance of the RHR system leakage test. The performance of the RHR system leakage test on a refueling basis instead of every 12 months cannot create a new or different kind of accident.

Therefore, for the reasons discussed above, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated, and the accident analyses presented in the Updated Safety Analysis Report [USAR] will remain bounding.

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

The performance of the RHR system leakage test at power is more complex than performing the test during refueling shutdown. It is preferable, from an RHR system reliability and plant safety standpoint, to perform the test during refueling shutdown when the RHR system is already operating and when no changes to the RHR system configuration are required. Any possible increase in the risk to the public health and safety incurred by extending the RHR leak test interval from 12 months to refueling shutdown will be off-set by the reduction in risk obtained by not performing the RHR system leakage test during power operation.

The extension of the test interval would mean that possible RHR leakage could exist undetected for a longer period than allowed by the current Technical Specifications. However, the possible consequences of leakage from the RHR system outside containment are minor relative to those of the design basis accident. In addition, it is probable that RHR system leakage would be identified during the normal quarterly functional testing and inspection of the RHR system.

Based on the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety as defined in the USAR or the Technical Specification Bases.

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.

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

Date of amendment request: December 23, 1994.

Description of amendment request: The proposed amendment to the Technical Specifications revises the surveillance requirement to perform a visual inspection of containment areas affected by containment entry when containment integrity is established. It is consistent with Item 7.5 of Generic Letter 93–05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation.''

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

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

The proposed change does not alter the assumptions, design parameters or results of Updated Final Safety Analysis Report (UFSAR) accidents analyzed. The proposed change does not involve a hardware change, a change to the operation of any systems or components, or a change to any existing structures. The proposed change leads to a reduction in radiation exposure to plant personnel and the elimination of an unnecessary burden on plant staff. The revised visual inspection practice will not increase the probability or consequences of an accident previously evaluated.

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

The proposed change does not modify equipment, affect system design bases or operability. This change does not alter parameters utilized in the analyzed accident scenarios. The proposed change in surveillance frequency is consistent with the guidance provided in GL 93–05. The performance of a visual inspection of containment areas affected by multiple containment entries on a daily bases [basis] and at the completion of the final entry when containment integrity is established will not create the possibility of a new or different kind of accident from those previously evaluated.

3. Does not involve a significant reduction in a margin of safety.

The proposed change only involves a decrease in surveillance frequency when multiple entries are made in a single day and does not alter the performance of the surveillance itself. System equipment and operation remains unchanged. Operability and reliability is still maintained by the required inspection. The adaptation of the proposed surveillance frequency does not involve a significant reduction in the margins 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: Salem Free Public Library, 112 West Broadway, Salem, New Jersey 08079.

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: December 16, 1994 (TS 94–06).

Description of amendment request: The proposed change would revise the auxiliary feedwater system technical specifications and associated Bases by incorporating the Westinghouse Standard Technical Specification limits and format, extending the limiting condition for operation to Mode 4, relaxing the achievement of hot shutdown from 6 hours to 12 hours, relaxing the verification of valve position surveillance frequency from 7 days to 31 days, and verifying the position of automatic valves every 31 days in lieu of valve manipulation.

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:

TVA has evaluated the proposed technical specification (TS) change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

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

The proposed TS change replaces SQN's auxiliary feedwater (AFW) system specification and the associated bases section with improved requirements that are modeled after the Westinghouse Standard (NUREG–1431) Technical Specification (STS). The proposed change is consistent with the STS for ensuring that three trains of AFW remain operable in Modes 1, 2, and 3. In addition, the proposed change provides a TS improvement by extending the limiting condition for operation (LCO) applicability to Mode 4. This LCO requirement for Mode 4 ensures that at least one motor-driven AFW pump remains operable when steam generators are being used for decay heat removal. The proposed 72 hour allowed outage time (for one inoperable train of AFW) is consistent with the STS and remains unchanged from SQN's current allowed outage time. One proposed change to relax shutdown requirements from 6 hours to 12 hours for achieving hot shutdown is considered to be acceptable. This relaxation is based on shutdown times contained in the STS and the operating experience to reach thus condition from full power in an orderly manner without challenging plant systems. The proposed surveillance requirements (SRs) provide test frequencies that are consistent with the STS and are based on operating experience and the design reliability of the equipment. The proposed relaxation in surveillance frequency from 7 days to 31 days for verifying valve position in the AFW flow path is considered acceptable based on existing procedural controls for valve configuration. The proposed change to include a STS SR for verifying automatic valves in the flow path are in their correct position every 31 days (in lieu of valve manipulation) is considered acceptable based on existing surveillance that verify proper actuation of SQN's automatic AFW valves.

The proposed changes provide TS improvements for SQN's AFW system that ensure the system operates within the bounds of SQN's AFW accident analysis as contained in the Final Safety Analysis Report (FSAR). This change does not involve a physical modification to SQN's AFW system. Accordingly, the proposed changes do not involve an 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 previously analyzed.

The proposed TS change incorporates requirements that bound the limiting design-basis accidents (DBAs) evaluated in SQN's FSAR. The TS bases have been revised to reflect the limiting DBAs and provide clarification with regard to the assumptions used in SQN's AFW accident analysis. No new event initiator has been created, not [sic] has any hardware been changed. This change does not involve a physical change to SQN's AFW system or any other system. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously analyzed.

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

TVA's proposed change replaces SQN's AFW system TS requirements with TS requirements adopted from the Westinghouse STS. Because the overall similarity in the requirements between SQN's current AFW specification and the STS version, the TS requirements remain essentially unchanged. The proposed 72-hour allowed outage time (for one inoperable train of AFW) is consistent with the STS and remains unchanged from SQN's current allowed outage time. One proposed change to relax shutdown requirements from 6 hours to 12 hours for achieving hot shutdown is considered to be acceptable. This relaxation is based on shutdown times contained in the STS and the operating experience to reach this condition from full power in an orderly manner without challenging plant systems. The proposed SRs provide test frequencies that are consistent with the STS and are based on operating experience and the design reliability of the equipment. The proposed relaxation in surveillance frequency from 7 days to 31 days for verifying valve position in the AFW flow path is considered acceptable based on existing procedural controls for valve configuration. The proposed relaxation in surveillance frequency from 7 days to 31 days for verifying valve position in the AFW flow path is considered acceptable based on existing procedural controls for valve configuration. The proposed change to include a STS SR for verifying automatic valves in the flow path are in their correct position every 31 days (in lieu of valve manipulation) is considered acceptable based on other existing surveillances that verify proper actuation of SQN's automatic AFW valves.

The proposed changes provide TS improvements for SQN's AFW System that ensure the system operates within the bounds of SQN's AFW accident analysis as contained in the FSAR. This change does not involve a physical modification to SQN AFW system. Accordingly, the margin of safety has not been reduced.

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

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11H, Knoxville, Tennessee 37902.

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: December 16, 1994.

Description of amendment request: The proposed license amendment would revise Technical Specification 6.3, ''Unit Staff Qualifications.'' Currently, the Technical Specifications require that the Operations Manager obtain a senior reactor operator (SRO) license by August 1995. A change is proposed to relieve the requirement for the Operations Manager to hold a Perry Nuclear Power Plant (PNPP) SRO license if an Operations section middle manager holds a PNPP SRO license.

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

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

The proposed change affects an administrative control, which was based on the guidance of ANSI N18.1–1971, ''Selection and Training of Nuclear Power Plant Personnel.'' ANSI N18.1–1971 recommended that the Operations Manager hold a senior reactor operator (SRO) license. The current guidance in Section 4.2.2 of ANSI/ANS–3.1–1993, ''American National Standard for Selection, Qualification, and Testing of Personnel for Nuclear Power Plants'' recommends, as one alternative, that the Operations Manager have plant operational knowledge consistent with the requirements of the Operations Manager's position, providing an Operations middle manager holds an SRO license. This individual (currently designated as the Operations Superintendent) would be required to meet the criteria for, and would have responsibilities as recommended in, ANSI/ANS–3.1–1993 for the Operations Middle Manager position. The proposed change is consistent with the recommendations of ANSI/ANS–3.1–1993.

The proposed change does not alter the design of any system, structure or component, nor does it change the way plant systems are operated. It does not reduce the knowledge, qualifications, or skills of licensed operators, and does not affect the way the Operations Section is managed by the Operations Manager. The Operations Manager will continue to maintain the effective performance of section personnel and ensure the plant is operated safely and in accordance with the requirements of the operating license. Additionally, the control room operators will continue to be supervised by the licensed senior operators such as the Unit Supervisors and the Shift Supervisors. For those areas of knowledge that require an SRO license, the Operations Superintendent will provide the appropriate technical guidance to the control room staff.

In summary, the proposed change does not affect the ability of the Operations Manager to provide the plant oversight required of the position. Thus, it does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to Technical Specification 6.3.1 does not affect the design or function of any plant system, structure, or component, nor does it change the way plant systems are operated. It does not affect the performance of NRC licensed operators. Operation of the plant in conformance with the Technical Specifications and other license requirements will continue to be supervised by personnel who hold an NRC SRO license. The proposed change to Technical Specifications 6.3.1 ensures that either the Operations Manager or Operations Superintendent will be a knowledgeable and qualified individual by requiring one of the individuals to hold an SRO license for PNPP. Based on the above, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change involves an administrative control which is not related to the margin of safety as defined in the Technical Specifications. The proposed change provides an alternative which ensures that the level of knowledge and experience required of an individual who fills the Operations Manager position is acceptable. The proposed change does not affect the conservative manner in which the plant is operated. The control room operators will continue to be supervised by personnel who hold an SRO license. Thus, 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: Perry Public Library, 3753 Main Street, Perry, Ohio 44081.

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

NRC Project Director: Leif J. Norrholm.

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

Date of amendment request: December 21, 1994.

Description of amendment request: The proposed license amendment would revise Technical Specification 3/4.3.7.7, ''Traversing In-Core Probe System,'' and its Bases to allow the use of substitute data generated from the process computer, normalized with available operating measurements, to replace data from inoperable local power range monitor (LPRM) strings for up to 10 LPRM strings.

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 TIP [traversing in-core probe] system is not used to prevent or mitigate the consequences of any previously analyzed accident or transient. No assumptions are made in any accident analysis relative to the operation of the TIP system. No other safety related system is affected by this change.

The use of substitute values from calculations performed by the on-line computer core monitoring system does not affect the consequences of plant transients previously evaluated in the USAR [Updated Safety Analysis Report] because the total core TIP reading (nodal power) uncertainty remains less than 8.7%. Thus, the MCPR [minimum critical power ratio] safety limit is not affected.

2. The proposed change does not create the possibility of a new or different kind of

accident from any accident previously evaluated.

The proposed change does not involve the installation of any new equipment or the modification of any equipment designed to prevent or mitigate the consequences of accidents or transients. Therefore, the change has no effect on any accident initiator, and no new or different type of accidents are postulated to occur.

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

The total core TIP reading uncertainties will remain within the assumptions of the licensing basis; thus, the margin of safety to the MCPR safety limits is not reduced. The ability of the computer to accurately represent nodal powers in the reactor core is not compromised. The ability of the computer to accurately predict the LHGR [linear heat generation rate], APLHGR [average planar linear heat generation rate], MCPR, and its ability to provide for LPRM calibration, are not compromised. Therefore, the margin of safety is not significantly reduced.

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

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, Ohio 44081.

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

NRC Project Director: Leif J. Norrholm.

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

Date of amendment request: December 6, 1994.

Brief description of amendments: The proposed amendment would revise Technical Specifications to allow appropriate remedial action for high particulate levels in the diesel generator fuel oil inventory and other out-of-limit properties in new diesel generator fuel oil that has been added to the existing diesel generator fuel oil storage inventory.

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

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

The proposed changes allow 7 days to correct particulate contamination in the stored fuel oil for the diesel generators and 30 days to confirm or restore the adequacy of the stored fuel oil if certain properties of new fuel that has been added to the fuel oil storage inventory have been discovered to exceed the specified values. These changes do not affect plant operations and the only equipment affected are the diesel generators. The ability of the diesel generators to provide electrical power when needed is directly dependent upon, in part, having fuel oil of adequate quality. The only accident which is potentially initiated by a diesel generator failure is the station blackout event. The mitigation of many accidents is dependent upon the availability of at least one train of electrical power from an emergency diesel generator (EDG). With the proposed changes, the fuel oil should continue to have sufficient quality to assure the operability of the diesel generators until the particulate and other properties are returned to within limits. This is due in part to the existing fuel oil quality requirements that are more stringent than the vendor requires for the EDG to operate and the system of filters installed to insure good quality fuel actually reaches the EDG. Even though the margin provided in the quality of the fuel oil may be affected (see the response to question 3 below), adequate fuel oil quality is being maintained to assure the operability of the diesel generators and therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

There are no hardware changes and no changes in system operations involved. These changes only affect the quality of the stored fuel oil for the diesel generators. The availability of a diesel generator has been addressed by the CPSES [Comanche Peak Steam Electric Station] design and in particular by the analysis of the station blackout event. These changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety of interest for these changes is the quality of the stored fuel oil for the diesel generators as compared to minimum quality which will support the diesel generators ability to supply electrical power when needed. Particulate contamination increases slowly over a period of time due to the chemical breakdown of the fuel oil (or its additives or the surfaces on the tanks themselves) or due to the introduction of foreign material during refueling activities. When considered with the fact that the existing limitation of 10 mg/L was developed for engines which require much cleaner fuel oil (aircraft engines) and that the CPSES diesel engines have in line duplex fuel oil filters which can be switched while the engine is operating, the 7 days which are being provided to restore the particulate levels do not involve a significant reduction in the margin of safety. The levels of particulate are expected to not exceed the specified value by a significant amount and the specified value is already quite conservative. Seven days is a reasonable time period in which to restore the parameter but is short enough to ensure that the contamination values do not exceed the vendors recommended fuel oil tolerances required for the EDGs to run. In a similar manner, the properties of the new fuel oil that has been added to the fuel oil storage inventory are not expected to deviate significantly from the allowed values. The testing for gravity, viscosity, flash point, clarity, water and sediment prior to adding the new fuel oil provides adequate assurance that the stored fuel oil will be of sufficient quality to support diesel generator operation. The quality of the stored fuel oil is further protected from problems being introduced by new fuel oil that has been added to the fuel oil storage inventory by the fact that the new fuel oil is generally diluted by a factor of four or more when it is added to the storage tanks by the fuel oil that is already in the tanks. Allowing 30 days to confirm or restore the properties of the stored fuel oil when a sample of new fuel that has been added to the fuel oil storage inventory has properties which exceed their specified values 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: University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, Texas 76019.

Attorney for licensee: George L. Edgar, Esq., Newman and Holtzinger, 1615 L Street, N.W., Suite 1000, Washington, D.C. 20036.

NRC Project Director: William D. Beckner.

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

Date of amendment request: December 7, 1994.

Brief description of amendments: The proposed amendment to the technical specifications (TSs) would: (1) revise the Comanche Peak Steam Electric Station (CPSES), Technical Specification Limiting Condition for Operation (LCO) for the main steam isolation valves (MSIVs) to increase the allowed outage time (AOT) in Mode 1; (2) relocate the MSIVs full closure time requirement to a program administratively controlled by the TS; and (3) revise the associated Bases to adopt the expanded Bases format adding information specific to CPSES.

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

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

The proposed changes are to (1) revise the CPSES Technical Specification Limiting Condition for Operation (LCO) for the MSIVs to increase the Allowed Outage Time (AOT) from 4 hours to 8 hours in Mode 1; (2) modify the Mode 2 and 3 Action Statement to better reflect the safety significance of these valves by requiring that the valves be closed within 8 hours and verified at least every 7 days; (3) relocate the MSIVs full closure time requirement to a program administratively controlled by the TS; and (4) revise the associated Bases to adopt the expanded Bases format adding information specific to CPSES.

The revision of the CPSES Technical Specification Limiting Condition For Operation (LCO) for the MSIVs to increase the Allowed Outage Time (AOT) from 4 hours to 8 hours in Mode 1 only affects the time that a condition can exist and as such does not affect any of the conditions that could initiate an accident; therefore the probability of an accident is not affected. Likewise, no new conditions are created that would affect the analyses of any accident; therefore the consequences of the accidents postulated for CPSES are not affected.

Modifying the Mode 2 and 3 Action Statement to better reflect the safety significance of these valves by requiring that the valves be closed within 8 hours and verified at least every 7 days provides clarity and adds a new verification requirement. Again no new plant conditions are established, time limits and verification requirements are merely being established; therefore, no accident initiators are affected and there is no impact on the probability of any accident. Likewise no conditions are being altered which affect the analyses of any accidents which are postulated at CPSES and thus the consequences of those accidents are unaffected.

Relocating the MSIVs full closure time requirement to a program administratively controlled by the TS is an administrative change only. It has no impact on actual plant operation and thus there is no impact on the probability of any accident or on the consequences of any accident.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated? None of the changes in this request affect plant design or create new operating configurations. The only things affected are the times that certain conditions are allowed, how soon actions need be performed, how often to verify conditions and the administrative location of certain requirements. These items do not create the possibility of a new type or different kind of accident.

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

The Technical Specifications LCOs ensure that the assumptions of the safety analyses are preserved. There are no substantive changes to the LCO; therefore, the safety analyses are unaffected and there is no affect on the margin of safety.

Revising the CPSES Technical Specification Limiting Condition For Operation (LCO) for the MSIVs to increase the Allowed Outage Time (AOT) from 4 hours to 8 hours in Mode 1 allows the unit to operate with an inoperable MSIV for a longer period of time. Although the unavailability of equipment required to mitigate or assess the consequence of an accident is increased, a more reasonable completion time is provided to diagnose the problem, mobilize the corrective action, obtain administrative clearances, complete the maintenance, restore the valve to an operable condition, and perform post-maintenance verification, where appropriate. The additional time would reduce the probability of unnecessary plant transients and plant shutdowns, thus improving plant safety and increasing plant availability, while a qualitative assessment has concluded that the impact on Core Damage Frequency is negligible. TU Electric has concluded based on the discussion above that there is no significant impact on the overall margin of safety due to this change.

Modifying the Mode 2 and 3 Action Statement to better reflect the safety significance of these valves by requiring that the valves be closed within 8 hours and verified at least every 7 days is primarily a clarification and a new verification requirement. Specifying that an inoperable valve be closed within 8 hours makes the requirement specific where no time limit was provided before. The 8 hours specified is the same as is allowed in Mode 1 which was qualitatively assessed as noted above and thus is a logical limitation. The new requirement to verify the valves closed on a periodic basis will increase assurance that the valves remain closed and will thus enhance the margin of safety. Overall, TU Electric concludes that these Mode 2 and 3 changes do not significantly affect the margin of safety.

Relocating the MSIVs full closure time requirement to a program administratively controlled by the TS is an administrative change only. There is no impact on the margin of safety.

Revising the associated Bases to adopt the expanded Bases format adding information specific to CPSES enhances the useability of the Technical Specification. Overall, this is considered an improvement which will benefit both the operators and support personnel. There is no significant impact on the margin of safety and if there is an impact, it improves the margin by providing easy access to support information.

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

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

Attorney for licensee: George L. Edgar, Esq., Newman and Holtzinger, 1615 L Street, N.W., Suite 1000, Washington, D.C. 20036.

NRC Project Director: William D. Beckner.

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

Date of amendment request: December 19, 1994.

Brief description of amendments: The proposed changes to the Technical Specification Action Statements of Tables 3.3–1 and 3.3–2 would allow testing of the reactor protective system (RPS) and the engineered safety features actuation system (ESFAS) with the channel under test in bypass.

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

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

The proposed changes will revise those Action Statements which limit the use of bypass while testing for Reactor Protection System (RPS) and Engineered Safety Feature Actuation System (ESFAS) functions. The Actions Statements concern testing with a channel inoperable and will be revised to allow testing with either the inoperable channel or the channel being tested (but not both) placed in bypass.

Testing in a bypass condition when all channels are operable will not introduce new operating configurations. The number [of] available channels with one channel in bypass for testing will remain the same as the minimum number of channels and is the same as the number of channels available when testing in trip. The number of channels to trip will be unchanged when testing in bypass while the number of channels to trip is reduced to one when testing in trip. Although there may be a sight [slight] increase in possibility that the failure of a channel could prevent the actuation of a function (because testing in bypass could result in two-out-of-two logic while testing in trip would have resulted in one-out-of-two logic), testing in bypass will reduce the vulnerability to inadvertent actuation of a function while maintaining the normal channels to trip and the minimum channels operable requirements per the current technical specifications. Overall TU Electric concludes (and WCAP–10271 with its associate SER from the NRC supports) that testing in bypass when all channel [s] are operable does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Testing in bypass with one channel inoperable will not introduce new configurations. The current Actions Statements for ESFAS already allow testing in bypass if one channel is inoperable. Under the current Technical Specifications for an RPS function, an inoperable channel is placed in bypass (via leads and jumpers) while surveillance testing another channel (the channel under test is placed in trip). Under the proposed changes, either the inoperable channel or the channel being tested may be bypassed.

In either case, the result is one channel in bypass and the other in trip, which leaves one-out-of-two operable channels to initiate the protective function (if the initial logic was two-out-of-four) or one-out-of-one operable channels to initiate the protective function (if the initial logic was two-out-of-three). Thus, testing in bypass with one channel inoperable does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed technical specification changes will also allow certain ESFAS functions to be tested with an inoperable channel in bypass and the channel being tested in trip. The current technical specifications require that the inoperable channel be in trip and that the channel being tested be in bypass. Per the same logic provided above on testing in bypass with an inoperable channel, this change has no impact on the capability of the system to respond to plant conditions and does increase the potential for inadvertent actuation of a function.

In summary, the proposed changes to the technical specifications and testing in bypass do not increase the probability or consequences of an accident previously evaluated.

(2) Do the proposed changes create the possibility of a new or different type of accident from any accident previously evaluated?

No new operating configurations and no new failure modes are being introduced by testing in bypass or by the proposed technical specification changes; therefore, no new or different type of accident from any accident previously evaluated is being created.

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

Testing in bypass does not affect accident configurations, sequences, or response scenarios as modeled in the safety analyses. Testing or maintenance in a bypass configuration does not cause any design or analysis acceptance criteria to be exceeded, nor does it affect the integrity of the fission product barriers. The severity of any accident previously evaluated is not increased. Bypass testing does not affect the functional integrity of the Reactor Protection System (RPS) or the Engineered Safety Features Actuation System (ESFAS). Bypass testing and the proposed technical specification changes do not involve a significant reduction in the margin of safety.

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

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

Attorney for licensee: George L. Edgar, Esq., Newman and Holtzinger, 1615 L Street, N.W., Suite 1000, Washington, D.C. 20036.

NRC Project Director: William D. Beckner.

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

Date of amendment request: December 30, 1994

Brief description of amendments: The proposed amendments would revise the technical specification for fuel storage to authorize use of the high density fuel storage racks, to increase the spent fuel storage capacity, and to adopt the wording, content, and format of the Improved Standard Technical Specifications.

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

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

This proposed license amendment includes changes which clarify the Technical Specifications, identify existing licensing basis criteria, revise the wording and format to be consistent with the Improved Standard Technical Specifications (NUREG–1431), and provide the criteria for acceptable fuel storage in high density racks. The clarification and the revised wording and format are purely administrative changes and have no impact on the probability or consequences of an accident. The criteria for acceptable fuel storage in the high density racks are discussed below.

The high density racks differ from the low density racks in that the center to center storage cell spacing is decreased from a nominal 16 inches to a nominal 9 inches and the high density racks are free standing whereas the low density racks are bolted to the pool. The allowed storage pattern in the high density racks results in a nominal 12.7 inch center to center spacing (measured diagonally) with a two out of four storage pattern (high density (2/4)). Administrative controls are used to maintain the specified storage patterns and to assure storage of a fuel assembly in a proper location based on initial U–235 enrichment and burnup. The increased storage capacity results in added weight in the pools and additional heat loads.

The only potential impact on the probability of an accident concerns the potential insertion of a fuel assembly in an incorrect location in the high density racks. TU Electric has used administrative controls to move fuel assemblies from location to location since the initial receipt of fuel on site. Through receipt of fuel for two initial core loads and four refueling outages (each of which includes a complete core offload), TU Electric has not inserted a fuel assembly into an improper location. This record demonstrates the adequacy of the administrative controls in place and confirms that the use of such administrative controls will not involve a significant increase in the probability of an accident previously evaluated.

The consequences of all of these changes have been assessed and the current acceptance criteria in the licensing basis of CPSES will continue to be met. The nuclear criticality, thermal-hydraulic, mechanical, material and structural designs will accommodate these changes. Potentially affected analyses, including a dropped spent fuel assembly, a loss of spent fuel pool cooling, a seismic event, and a fuel assembly placed in a location other than a prescribed location, continue to satisfy the CPSES licensing basis acceptance criteria. The analysis methods used by TU Electric are consistent with methods used by TU Electric in the past or methods used elsewhere in the industry and accepted by the NRC.

Based on the acceptability of the methodology used and compliance with the current CPSES licensing basis, TU Electric concludes that the use of the high density racks and the increase in storage capacity do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The administrative changes to the Technical Specifications have no impact on plant hardware or operations and therefore cannot create a new or different kind of an accident.

The spacing changes between fuel assemblies, the administrative controls, the storage limitations, and the increased storage capacity do not generate new failure modes that could create a new or different kind of an accident. The change from bolted low density racks to free standing high density racks will not create the possibility of a new or different kind of an accident. Free standing racks have been commonly used at nuclear power plants to provide for high density storage of spent fuel, and their use does not entail any unproven or unusual design or technology. In this regard, a number of plants have previously changed from bolted or restrained racks to free standing racks, including Millstone 1 (amendment dated November 27, 1989) and San Onofre 2 and 3 (amendment dated May 1, 1990), and such changes have not been classified as involving a significant hazards consideration. Furthermore, CPSES is not located in an area subject to severe seismic events. A seismic event at CPSES would result in little movement of the free standing racks and would not cause the high density racks to collide with each other or the spent fuel pool walls. Therefore, use of the free standing high density racks would not create the possibility of a new or different kind of an accident.

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

The proposed administrative changes to the Technical Specifications have no impact on any acceptance criteria, plant operations or the actual failure of any systems, components or structure; therefore these administrative changes have no impact on the margin of safety.

The NRC guidance [Nuclear Regulatory Commission, Letter to all Power Reactor Licensees, from B. K. Grimes, April 14, 1978, ''OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications,'' as amended by the NRC Letter dated January 18, 1979] has established that an evaluation of margin of safety should address the following areas:

(1) Nuclear criticality considerations.

(2) Thermal-Hydraulic considerations.

(3) Mechanical, material and structural consideration.

The established acceptance criterion for criticality is that the neutron multiplication factor in the spent fuel pool storage racks shall be less than or equal to 0.95, including uncertainties, under all conditions. The k_{eff} for the high density racks for CPSES is always less than 0.95, including uncertainties at a 95/95 probability confidence level. Because the existing acceptance criterion is shown to be satisfied, the high density racks do not involve a significant reduction in the margin of safety with respect to criticality considerations.

The thermal-hydraulic evaluation demonstrates that the temperature margin of safety will be maintained. Re-evaluation of the spent fuel pool cooling system for the increased heat loads shows, with minor modifications, that the spent fuel cooling system will maintain the abnormal maximum temperature of the spent fuel pool water within the limits of the existing licensing basis (i.e., below 212 °F). Additionally, it shows that, with minor modifications, the normal maximum temperature will be within the existing design basis temperatures for the high density racks, liner, structure, and cooling system and will not have any significant impact on the spent fuel pool demineralizers. Thus, the existing licensing basis remains valid, and there is no significant reduction in the margin of safety for the thermal-hydraulic design or spent fuel cooling.

The main safety function of the spent fuel pool and the high density racks is to maintain the spent fuel assemblies in a safe configuration through normal and abnormal operating conditions. The design basis floor responses of the Fuel Building were confirmed to be adequate and conservative and the floor loading will not exceed the capacity of the Fuel Building. The high density rack materials used are compatible with the spent fuel pool and the spent fuel assemblies. The structural considerations of the high density racks maintain margin of safety against tilting and deflection or movement, such that the high density racks do not impact each other or the pool walls, damage spent fuel assemblies, or cause criticality concerns. Thus, the margin of safety with respect to mechanical, material and structural considerations are not significantly reduced by the use of the high density racks.

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

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

Attorney for licensee: George L. Edgar, Esq., Newman and Holtzinger, 1615 L Street, N.W., Suite 1000, Washington, DC 20036.

NRC Project Director: William D. Beckner.

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

Date of amendment request: December 9, 1994.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 4.3.2.2, 4.7.1.2.1, and the Bases for Specification 3/4.7.1.2. The changes would decrease the frequency of testing auxiliary feedwater pumps, provide consistent testing requirements for the steam turbine-driven auxiliary feedwater pump, and clarify performance parameters in the Bases.

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

The proposed revision does not involve a significant hazards consideration because operation of Callaway Plant with this change would not:

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

The Callaway Final Safety Analysis Report has been reviewed and been found to be unaffected by these proposed changes. The changes proposed by this Technical Specification amendment do not affect the performance parameters of the Auxiliary Feedwater System (AFWS). The changes proposed involve a decrease in the frequency of pump testing from once per 31 days to once per 92 days as recommended by NRC Generic Letter 93–05 and reflected in NUREG–1431 (T/S 4.7.1.2.1.a). This change will decrease the out-of-service time of the AFWS due to testing. This change will also decrease the number of component manipulations performed on the system and will therefore decrease the probability of a restoration error rendering the system incapable of performing its intended function.

The pumps will be required to meet the same acceptance criteria and will continue to be monitored as required by ASME Section XI. As stated earlier, the overall effect is a slight decrease in the CDF for Callaway. These proposed changes will also eliminate an inconsistency among Specifications 4.7.1.2.1.b.2 and 4.3.2.2 and Specification 4.7.1.2.1.a.2 regarding an exception to Specification 4.0.4 for entry into Mode 3 for the TDAFP. The methodology and acceptance criteria of surveillance testing will not be changed. The ability of the AFWS to perform its intended function during accident conditions will continue to be demonstrated via surveillance testing. The proposed changes to the Technical Specifications do not affect any accident initiators for any accident evaluated in the Final Safety Analysis Report (FSAR). The Bases changes are corrections to errors which have no effect on any accident initiators nor equipment failure modes.

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

The proposed Technical Specification changes do not modify any equipment nor create any potential accident initiators. The proposed change herein of potential interest is the exception to Specification 4.0.4 for entry into Mode 3 for TDAFP response time testing and auto-start testing. This allowance is already recognized via Specification 4.7.1.2.1.a.2 and NUREG–1431, Standard Technical Specifications-Westinghouse Plants.

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

The Bases for Specification 3/4.7.1.2 are to be clarified to correctly state the design flow and pressure parameters for the AFWS. No plant design changes are involved in any of the proposed changes and the method and manner of plant operation remain the same. The specific surveillance test methodology and acceptance criteria remain unchanged.

As discussed above, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated or create the possibility of a new or different kind of accident from any previously evaluated. These changes do not result in a significant reduction in a margin of safety. Therefore, it has been determined that the proposed changes do not involve a significant hazards consideration.

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: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, DC 20037.

NRC Project Director: Leif J. Norrholm.

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

Date of amendment request: December 9, 1994, as supplemented on December 22, 1994.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Surveillance Requirement 4.8.1.1.2f.7 to remove the requirement to perform the hot restart test within 5 minutes of completing the 24-hour endurance test and place that requirement in a separate TS.

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

The proposed revision does not involve a significant hazards consideration because operation of Callaway Plant with this change would not:

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

The proposed revision to the T/S will not adversely impact plant safety since the requirement to perform the hot restart test will still be implemented via a separate surveillance requirement that demonstrates the hot restart functional capability of the diesel generators.

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

There are no design changes being made that would create a new type of accident or malfunction and the method and manner of plant operation remain unchanged. The performance capability of the emergency diesel generators will not be affected. The verification of the hot restart capability of the diesel generators will still be performed, only the timing of the performance will be changed to give plant operators added flexibility and prevent critical path complications during outages.

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

There are no changes being made to the safety limits or safety system settings that would adversely impact plant safety. The diesel generators will still perform their intended safety function following a loss of offsite power, to achieve and maintain the plant in a safe shutdown condition.

Based on the above discussions, it has been determined that the requested Technical Specification change does not involve a significant increase in the probability or consequences of an accident or create the possibility of a new or different kind of accident or condition over previous evaluations; or involve a significant reduction in a margin of safety. Therefore, the requested license amendment does not involve a significant hazards consideration.

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: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

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

NRC Project Director: Leif J. Norrholm.

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

Date of amendment request: December 14, 1994.

Description of amendment request: The proposed amendment would revise instrument identification for low reactor pressure instrument trip cards in emergency core cooling system (ECCS) actuation to reflect a design change to be installed during the 1995 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:

1. The proposed change to the identification numbers for certain reactor pressure instrumentation as included in the Technical Specifications for ECCS Actuation Instrumentation is only necessary because the specific identification numbers (Tag Nos.) have been listed in the [***]. This is considered an administrative type change. Acceptable measurement of Low Reactor Pressure is still assured. All automatic control or trip functions will continue to be provided.

The proposed change does not result in any function or setpoint change. The hardware changes which have resulted in a need to change the Technical Specifications have removed instrumentation no longer required to be installed in the circuitry for measuring ECCS Low Reactor Pressure. The existing logic for Low Reactor Pressure will remain the same. The only change applicable to implementation of the design modification is the use of different trip cards to provide the trip function for ECCS Low Reactor Pressure.

The requested change to ECCS Actuation Instrumentation Tables does not impact any FSAR [Final Safety Analysis Report] safety analysis involving the ECCS or Protection Systems. These measurement functions are not contributors to the initiation of accidents.

The change in instrument Tag Nos. on Tables 3.2.1 and 4.2.1 will have no affect on any safety limit setting or plant system operation and, therefore, does not modify or add any initiating parameters that would significantly increase the probability or consequences of any previously analyzed accident.

The administrative change to correct a typographical error on Table 4.2.1 will have no affect on plant hardware, plant design, safety limit setting or plant system operation and, therefore, does not modify or add any initiating parameters that would significantly increase the probability or consequences of any previously analyzed accident.

Therefore, it is concluded that there is not a significant increase in the probability or consequence of an accident previously evaluated.

2. The proposal to change instrument Tag Nos. does not result in any function changes or changes to Technical Specification requirements pertaining to these functions.

The proposed change does not involve any change in Technical Specification trip setpoints, plant operation, redundancy, protective function or design basis of the plant. There is no impact on any existing safety analysis or safety design limits. Low Reactor Pressure instrumentation functions do not initiate nuclear system parameter variations which are considered potential initiating causes of threats to the fuel and the nuclear system process barrier or that would create any new or different kind of accident.

As discussed above, the proposed administrative change only corrects a typographical error concerning equipment identification numbers. This change does not affect any equipment and it does not involve any potential initiating events that would create any new or different kind of accident.

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

3. The proposal to change the identification numbers for certain reactor pressure instrumentation as included in the Technical Specifications for ECCS Actuation Instrumentation does not affect any existing safety margins. The change by itself is administrative. The hardware changes which have resulted in a need to change the Technical Specifications have been reviewed per 10 CFR 50.59(a)(2) and determined to not constitute an unreviewed safety question.

The change in Tag Nos. or the change in the instrumentation used to measure low reactor pressure does not preclude the ability of the Core Spray (CS) or Low Pressure Coolant Injection (LPCI) Systems to perform their safety function to mitigate the consequences of accidents or of any other safety system to accomplish its safety functions. Proper post-accident ECCS functioning will still be provided by safety class instruments used to measure reactor pressure.

The change to instrument Tag Nos. as listed in the Technical Specifications has no affect on the bases of Protective Instrumentation which is to operate to initiate required system protective actions. The changes to be implemented which have resulted in a need to change the Technical Specifications will actually improve the accuracy of reactor pressure measuring loops.

[***]

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: Brooks Memorial Library, 224 Main Street, Brattleboro, Vermont 05301.

Attorney for licensee: John A. Ritsher, Esquire, Ropes and Gray, One International Place, Boston, Massachusetts 02110–2624.

NRC Project Director: Walter R. Butler.

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

Date of amendment request: December 22, 1994.

Description of amendment request: The proposed amendment would modify Point Beach Nuclear Plant Technical Specification (TS) Section 15.3.3, ''Emergency Core Cooling System, Auxiliary Cooling Systems, Air Recirculation Fan Coolers, and Containment Spray,'' TS Section 15.3.4, ''Steam and Power Conversion System,'' TS Section 15.3.5, ''Instrumentation System,'' TS Section 15.3.7, ''Auxiliary Electrical Systems,'' TS Section 15.3.14, ''Fire Protection System,'' and TS Section 15.4.1, ''Operation Safety Review.'' The modifications would delete obsolete TSs, would provide spring 1995 outage-specific TSs as part of the ongoing diesel upgrade project, would update several TSs to be consistent with the upgrade project design changes, and would change one monthly testing requirement. In addition, the bases for Section TS 15.3.7 would be modified to be consistent with the proposed TS changes.

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

In accordance with the requirements of 10 CFR 50.91(a), Wisconsin Electric Power Company (Licensee) has evaluated the proposed changes against the standards of 10 CFR 50.92 and has determined that the operation of Point Beach Nuclear Plant, Units 1 and 2, in accordance with the proposed amendments [sic] does not present a significant hazards consideration. The analysis of the requirements of 10 CFR 50.92 and the basis for this conclusion are as follows:

1. Operation of the facility under the proposed Technical Specifications will not create a significant increase in the probability or consequences of an accident previously evaluated.

The probabilities of accidents previously evaluated are based on the probability of initiating events for these accidents. Initiating events for accidents previously evaluated for Point Beach include: control rod withdrawal and drop, CVCS malfunction (Boron Dilution), startup of an inactive reactor coolant loop, reduction in feedwater enthalpy, excessive load increase, losses of reactor coolant flow, loss of external electrical load, loss of normal feedwater, loss of all AC power to the auxiliaries, turbine overspeed, fuel handling accidents, accidental releases of waste liquid or gas, steam generator tube rupture, steam pipe rupture, control rod ejection, and primary coolant system ruptures.

This license amendment request proposes to remove the specifications associated with the 4160 volt safeguards bus tie, add and modify specifications associated with the degraded and loss of voltage protection functions, and remove specifications and surveillance exceptions that are obsolete. The modifications being performed and the changes proposed by this license amendment request have been reviewed and we conclude that these changes do not increase the probability of any initiating event for accidents previously analyzed for Point Beach Nuclear Plant.

The consequences of the accidents previously evaluated in the PBNP FSAR are determined by the results of analyses that are based on initial conditions of the plant, the type of accident, transient response of the plant, and the operation and failure of equipment and systems. The changes proposed in this license amendment request provide appropriate limiting conditions for operation, action statements, allowable outage times, surveillances and bases for the Point Beach Nuclear Plant Technical Specifications.

The proposed specification that allows a Train A service water pump powered from the alternate shutdown system to be considered operable under the provisions of Technical Specification 15.3.0.c is appropriate to maintain operability of the service water system for the continued safe operation of Unit 2 under the applicable standby emergency power limiting condition for operation.

The modifications that are being performed have been designed and will be installed in accordance with the applicable design and installation requirements for Point Beach Nuclear Plant.

Therefore, this proposed license amendment does not affect the consequences of any accident previously evaluated in the Point Beach Nuclear Plant FSAR because the factors that are used to determine the consequences of accidents are not being changed.

2. Operation of this facility under the proposed Technical Specifications change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

New or different kinds of accidents can only be created by new or different accident initiators or sequences. New and different types of accidents (different from those that were originally analyzed for Point Beach) have been evaluated and incorporated into the licensing basis for Point Beach Nuclear Plant. Examples of different accidents that have been incorporated into the Point Beach Licensing basis include anticipated transients without scram and station blackout.

The modifications being performed and the changes proposed by this license amendment request have been reviewed and we conclude that these changes do not create any new or different accident initiators or sequences. Therefore, these modifications and proposed Technical Specification changes do not create the possibility of an accident of a different type than any previously evaluated in the Point Beach FSAR.

3. Operation of this facility under the proposed Technical Specifications change will not create a significant reduction in a margin of safety.

The margins of safety for Point Beach are based on the design and operation of the reactor and containment and the safety systems that provide their protection. The modifications that are being performed have been designed and will be installed in accordance with the applicable design and installation requirements for Point Beach Nuclear Plant.

The modification to change the loss of voltage protection function from 1-out-of-2 logic on each bus to 2-out-of-3 logic on each bus is an improvement over the original design, because with the new design an inadvertent trip of a single channel will not cause the protection actions. Also, when any single channel is taken out-of-service for testing, maintenance, or calibration it can be placed in the trip condition to allow actuation of the protection function by the trip of either of the remaining operable channels.

The Technical Specification change to allow an operating pump powered from alternate shutdown to be considered operable is justified because the pump is able to perform its safety function powered from the alternate shutdown power source. The alternate shutdown system is powered via offsite power or from the onsite gas turbine generator and is being considered a normal power supply for the service water pump.

The alternate shutdown system was installed to provide an alternate means of providing power to service water pumps, component cooling water pumps, and residual heat removal pumps for certain 10 CFR 50 Appendix R fire scenarios in which the normal power supplies for this equipment become inoperable. As such, the alternate shutdown system is a qualified alternate source of power for the service water pump.

Therefore, the margins of safety for Point Beach are not being reduced because the design and operation of the reactor and containment are not being changed and the safety systems that provide their protection that are being changed are being modified in accordance with the applicable design and installation requirements for the Point Beach Nuclear Plant.

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: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

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

NRC Project Director: Leif J. Norrholm.

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.

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

Date of application for amendments: October 31, 1994, supplemented by letter dated December 28, 1994.

Brief description of amendment requests: The proposed amendments would change the refueling machine overload cutoff limit from less than or equal to 1556 pounds to less than or equal to 1600 pounds. The change is a consequence of the fuel assembly weight increase which resulted from design and fabrication improvements.

Date of individual notice in **Federal Register**: January 6, 1995 (60 FR 2160).

Expiration date of individual notice: February 6, 1995.

Local Public Document Room location: Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: August 30, 1994.

Description of amendment request: The proposed amendment revises technical specifications to address the installation of two battery chargers on each vital 125 vdc power train in lieu of the ''swing'' battery charger that is currently used.

Date of individual notice in the **Federal Register**: January 17, 1995 (60 FR 3439).

Expiration date of individual notice: February 16, 1995.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 728011.

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

Date of amendment request: October 21, 1994.

Brief description of amendment request: The proposed amendment would add the Special Test Exception 3/4.10.6, ''Inservice Leak and Hydrostatic Testing,'' that allows the performance of pressure testing at reactor coolant temperature up to 212 °F while remaining in OPERATIONAL CONDITION 4. This special test exception would also require that certain OPERATIONAL CONDITION 3 Specifications for Secondary Containment Isolation, Secondary Containment Integrity and Standby Gas Treatment System operability be met. This change would also revise the Index, Table 1.2, ''OPERATIONAL CONDITIONS,'' and the Bases to incorporate the reference to the proposed special test exception.

Date of publication of individual notice in **Federal Register**: December 22, 1994 (59 FR 66057).

Expiration date of individual notice: January 23, 1995.

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

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

Date of application for amendment: December 8, 1994.

Brief description of amendment: The proposed amendment would revise Section 4.4 of the Indian Point 3 Technical Specifications. Specifically, TS 4.4.E.1 would be revised to allow a one-time extension to the 30-month interval requirement for leak rate testing of Residual Heat Removal (RHR) containment isolation valves AC–732, AC–741, AC–MOV–743, AC–MOV–744, and AC–MOV–1870. This one-time extension for leak rate testing of the RHR valves would be deferred until prior to return to power following the current outage, which is defined as prior to T_{avg} exceeding 350 °F.

Date of publication of individual notice in **Federal Register**: December 13, 1994 (59 FR 64224).

Expiration date of individual notice: January 12, 1995.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

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: December 16, 1994.

Brief description of amendments: This amendment would revise Technical Specifications regarding diesel generator surveillance requirements.

Date of publication of individual notice in the **Federal Register**: December 22, 1994 (59 FR 67350).

Expiration date of individual notice: January 23, 1995.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

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: January 3, 1995.

Brief description of amendments: The amendments add a permissive statement to Surveillance Requirement 4.9.7.1 that will allow the auxiliary building bridge crane interlocks and physical stops to be defeated during implementation of the spent fuel pool storage capacity increase modification.

Date of publication of individual notice in the **Federal Register**: January 9, 1995 (60 FR 2404).

Expiration date of individual notice: January 24, 1995.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennesee 37402.

Notice of Insurance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendments: September 23, 1994.

Brief description of amendments: The amendments revise the Unit 2 Shutdown AC Power Sources TSs to permit a one-time increase the allowed outage time (AOT) from 7 to 14 days for the dedicated Class IE emergency power system and the Unit 1 control room emergency ventilation system TSs to permit a one-time increase the AOT from 7 to 30 days. These one-time extensions are necessary to support modifications scheduled to be implemented during the upcoming 1995 Unit 2 refueling outage.

Date of issuance: January 11, 1995.

Effective date: As of the date of issuance to be implemented during the 1995 Unit 2 refueling outage.

Amendment Nos.: Unit 1–202 and Unit 2–180.

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

Date of initial notice in **Federal Register**: October 26, 1994 (59 FR 53835).

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated January 11, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 15, 1993, as supplemented February 17, 1994, February 25, 1994, and November 23, 1994.

Brief description of amendment: The amendment deletes Section 2.C.(8) of the Facility Operating License NPF–63, and deletes Attachment 1 to the License, in response to your request dated May 15, 1993, as supplemented February 17, 1994, February 25, 1994, and November 23, 1994

Date of issuance: January 12, 1995.

Effective date: January 12, 1995.

Amendment No. 53.

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

Date of initial notice in **Federal Register**: June 9, 1993 (58 FR 32378).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 12, 1995.

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. 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 application for amendments: June 3, 1994.

Brief description of amendments: The amendments revise Byron and Braidwood technical specifications (TSs) to reflect a primary-to-secondary leakage rate of 150 gallons per day through any one steam generator and to reflect an inservice inspection of a minimum of 20 percent of a random sample of the sleeves at the end-of-cycle. The amendment also adds a condition to the licenses to conduct additional corrosion testing to establish the design life for the sleeved tubes in the presence of a crevice. The revised TSs are more conservative than the previous TSs and were requested in order to increase the confidence in the ability of sleeves to maintain primary-to-secondary integrity.

Date of issuance: January 6, 1995.

Effective date: January 6, 1995.

Amendment Nos.: 67, 67, 57, and 57.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revised the operating licenses and TSs.

Date of initial notice in **Federal Register**: October 12, 1994 (59 FR 51613). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 6, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: For Byron, the Byron Public Library, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Township Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

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 application for amendments: November 7, 1994, as supplemented December 16, 1994.

Brief description of amendments: The amendments approve the use and storage of fuel with an enrichment not to exceed a nominal 5.0 weight percent U–235 in the spent fuel racks.

Date of issuance: January 20, 1995.

Effective date: January 20, 1995.

Amendment Nos.: 68, 68, 58, and 58.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 7, 1994 (59 FR 63115). The December 16, 1994, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in an Environmental Assessment dated January 13, 1995, and in a Safety Evaluation dated January 20, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: For Byron, the Byron Public Library, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Township Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

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

Date of application for amendments: January 10, 1994, as supplemented September 15, 1994, January 5 and 10, 1995.

Brief description of amendments: The amendments revise Technical Specification (TS) Table 2.2–1 and TS 4.2.5 to allow a change in the method for measuring reactor coolant system (RCS) flow rate from the calorimetric heat balance method to a method based on a calibration of the RCS cold leg elbow differential pressure taps.

Date of issuance: January 12, 1995.

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

Amendment Nos.: 153 and 135.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: February 16, 1994 (59 FR 7688). The September 15, 1994, January 5 and 10, 1995, letters provided clarifying information that did not change the scope of the January 10, 1994, application, the **Federal Register** Notice or the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 12, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223

Florida Power and Light Company, et al., Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of application for amendments: November 2, 1994.

Brief description of amendments: These amendments clarify the actions required in the event of inoperable equipment associated with containment depressurization and cooling systems, and provide consistency between Unit 1 and Unit 2 requirements.

Date of Issuance: January 18, 1995.

Effective Date: January 18, 1995.

Amendment Nos.: 131 and 70.

Facility Operating License Nos. DPR–67 and NPF–16: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 7, 1994 (59 FR 63122).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 18, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: April 19, 1994.

Brief description of amendments: These amendments consist of changes to the Technical Specifications relating to surveillance requirements for inservice inspection and testing programs.

Date of issuance: January 11, 1995.

Effective date: January 11, 1995.

Amendment Nos. 171 and 165.

Facility Operating Licenses Nos. DPR–31 and DPR–41: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: May 25, 1994 (59 FR 27054).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 11, 1995.

No significant hazards consideration comments received: No

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

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50–424, Vogtle Electric Generating Plant, Unit 1, Burke County, Georgia

Date of application for amendment: August 16, 1994.

Brief description of amendment: The amendment eliminated License Condition 2.C.(6) and the associated Attachment 1 of the license. License Condition 2.C.(6) referenced Attachment 1 which listed special diesel generator maintenance and surveillance requirements.

Date of issuance: January 20, 1995.

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

Amendment No.: 81.

Facility Operating License Nos. NPF–68: Amendment revised the Facility Operating License.

Date of initial notice in **Federal Register**: September 6, 1994 (59 FR 46071).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 20, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Burke County Library, 412 Fourth Street, Waynesboro, Georgia 30830.

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: March 31, 1994.

Brief description of amendments: The changes revised TS Table 3.7–1 by lowering the maximum allowable power range neutron flux high setpoint when one or more main steam safety valves (MSSVs) are inoperable. The changes also revised the Bases for TS 3/4.7.1.1

to include the Westinghouse algorithm for determining the new setpoint values.

Date of issuance: January 20, 1995.

Effective date: To be implemented within 30 days from date of issuance.

Amendment Nos.: 82 and 60.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 20, 1994 (59 FR 37071).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 20, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Burke County Library, 412 Fourth Street, Waynesboro, Georgia 30830.

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: May 20, 1994.

Brief description of amendments: The amendments relocate the heat flux hot channel factor, $F_Q(Z)$, penalty of 2 percent in specification 4.2.2.2.f to the cycle-specific Core Operating Limits Report (COLR) to allow for burnup-dependent values of the penalty in excess of 2 percent. This amendment also revises the reference in specification 6.8.1.6 to the Westinghouse $F_Q(Z)$ surveillance methodology in order to reflect Revision 1 of WCAP–10216–P, ''Relaxation of Constant Axial Offset Control—F_Q Surveillance Technical Specification,'' approved by the NRC on November 26, 1993.

Date of issuance: January 11, 1995.

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

Amendment Nos.: 79 and 58.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 20, 1994 (59 FR 37072). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 11, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Burke County Library, 412 Fourth Street, Waynesboro, Georgia 30830.

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: August 16, 1994.

Brief description of amendments: The amendments change Technical Specification 3/4.7.1.1 and its Bases regarding the setpoint tolerance for the main steam safety valves.

Date of issuance: January 12, 1995.

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

Amendment Nos.: 80 and 59.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 14, 1994 (59 FR 47168).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 12, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Burke County Library, 412 Fourth Street, Waynesboro, Georgia 30830.

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: July 29, 1994.

Brief description of amendment: The proposed amendment would revise the Technical Specifications by deleting reference to written relief from ASME Code requirements. The revised Technical Specifications refer to the applicable provision of NRC regulations concerning the ASME Code.

Date of issuance: January 6, 1995.

Effective date: January 6, 1995, to be implemented within 120 days.

Amendment No.: 206.

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

Date of initial notice in **Federal Register**: August 31, 1994 (59 FR 45026).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 6, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, SE., Cedar Rapids, Iowa 52401.

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

Date of application for amendment: March 15, 1994, as supplemented October 20, 1994.

Brief description of amendment: This amendment allows the use of integral fuel burnable absorbers as a method of controlling core excess reactivity and maintaining core power distribution within acceptable peaking limitations.

Date of issuance: January 17, 1995

Effective date: January 17, 1995.

Amendment No.: 145.

Facility Operating License No. DPR–36: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 28, 1994 (59 FR 22010).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 17, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, Maine 04578.

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

Date of application for amendment: October 5, 1994.

Brief description of amendment: The amendment revises the applicability requirements of Technical Specification (TS) 3.7.3 to require operability of the Control Room Outdoor Air Special Filter Train System in Operational Conditions 1, 2, 3 and ** rather than in all Operational Conditions and **. The applicability requirements for Action Statement b. of TS 3.7.3 and for the Radiation Monitoring Instrumentation required operable by TS Tables 3.3.7.1–1 and 4.3.7.1–1 are being changed in a similar manner. The amendment also adds a notation to Action Statement b.1. of TS 3.7.3 stating that the provisions of TS 3.0.4 are not applicable for entry into Operational Condition ** when one filter train is inoperable provided an operable filter train is in operation in the emergency pressurization mode of operation.

Date of issuance: January 18, 1995.

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

Amendment No.: 60.

Facility Operating License No. NPF–69: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: November 9, 1994 (59 FR 55874).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 18, 1995.

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.

Northeast Nuclear Energy Company, Docket No. 50–245, Millstone Nuclear Power Station, Unit 1, New London County, Connecticut

Date of application for amendment: October 4, 1994.

Brief description of amendment: The amendment relocates the primary containment isolation valve list from Technical Specification (TS) Section 3.7.D to the Millstone Unit 1 Technical Requirements Manual. This change is in accordance with the guidance of Generic Letter 91–08. The amendment also makes administrative and editorial changes to TS Sections 3.7.D and 4.7.D and makes changes to the associated bases.

Date of issuance: January 10, 1995.

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

Amendment No.: 78.

Facility Operating License No. DPR–21. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 23, 1994 (59 FR 60383)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 10, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

Northeast Nuclear Energy Company, Docket No. 50–245, Millstone Nuclear Power Station, Unit 1, New London County, Connecticut

Date of application for amendment: October 14, 1994.

Brief description of amendment: The amendment clarifies the low pressure coolant injection requirements as required by Technical Specification 4.5.A.2.

Date of issuance: January 9, 1995.

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

Amendment No.: 77.

Facility Operating License No. DPR–21. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 7, 1994 (59 FR 63125).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 9, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

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: September 9, 1994, with clarifying information provided by letter dated October 5, 1994.

Brief description of amendment: The amendment revises the Technical Specifications to modify surveillance requirements by increasing the acceptance criterion for the closure of the main steam isolation valves from 5 seconds to 10 seconds.

Date of issuance: January 10, 1995.

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

Amendment No.: 101.

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

Date of initial notice in **Federal Register**: September 19, 1994 (59 FR 47960). The October 5, 1994, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated Janaury 10, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

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 application for amendments: October 17, 1994, as supplemented October 27, 1994.

Brief description of amendments: The amendments revise the Prairie Island Nuclear Generating Plant Technical Specifications to change the submittal frequency of the Radioactive Effluent Release Report from semiannual to annual in accordance with 10 CFR Part 50.36a.

Date of issuance: January 11, 1995.

Effective date: January 11, 1995, with full implementation within 30 days.

Amendment Nos.: 114 and 107.

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

Date of initial notice in **Federal Register**: December 7, 1994 (59 FR 63125) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 11, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 27, 1994.

Brief description of amendments: These amendments revise the Technical Specifications (TS) definition of ''Core Alteration'' to conform to the definition approved by the staff for the current boiling water reactor (BWR) improved TS in NUREG–1433, ''Standard Technical Specifications, General Electric Plants, BWR/4.''

Date of issuance: January 3, 1995.

Effective date: January 3, 1995.

Amendment Nos.: 138 and 108.

Facility Operating License Nos. NPF–14 and NPF–22. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 14, 1994 (59 FR 47177).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 3, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: August 22, 1994.

Brief description of amendments: These amendments change Technical Specifications 3/4.1.3 to: (1) Extend the scram discharge volume (SDV) vent or drain valve restoration time from the current time period of 24 hours to 7 days; (2) permit the SDV vent and drain valves operability check to be performed at shutdown conditions instead of at-least-once-per-18-months; and (3) delete the SDV float switch response surveillance requirement.

Date of issuance: January 9, 1995.

Effective date: January 9, 1995.

Amendment Nos.: 139 and 109.

Facility Operating License Nos. NPF–14 and NPF–22. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 28, 1994 (59 FR 49433).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 9, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: September 26, 1994.

Brief description of amendments: The amendments change the Technical Specifications for each of the units to remove the requirement for the average power range monitors (APRMs) to be operable while the plant is in Operational Condition 5, refueling status. However, the amendment does not change the requirement for the APRMs to be operable when the reactor mode switch is in Startup during a shutdown margin demonstration.

Date of issuance: January 9, 1995.

Effective date: January 9, 1995.

Amendment Nos.: 140 and 110.

Facility Operating License Nos. NPF–14 and NPF–22. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 9, 1994 (59 FR 55880).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 9, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: August 25, 1993, as supplemented by letter dated August 4, 1994.

Brief description of amendments: These amendments modify Technical Specification (TS) Section 3.3.7.8.2 and associated Bases 3/4.3.7.8 regarding the Main Control Room (MCR) toxic gas detection system. The TS change reflects the implementation of a modification designed to eliminate spurious high toxic gas concentration alarms received by the MCR.

Date of issuance: January 19, 1995.

Effective date: January 19, 1995.

Amendment Nos. 84 and 45.

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

Date of initial notice in **Federal Register**: September 29, 1993 (58 FR 50971).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 19, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 20, 1994, as supplemented September 23, 1994.

Brief description of amendments: The amendments would raise the Steam Leakage Detection system set-points that isolate the High Pressure Coolant Injection System (HPCI) and Reactor Core Isolation Cooling (RCIC) system equipment on high equipment room temperature and high delta temperature. The amendments are supported by a Limerick Generating Station modification to increase the environmental qualifications limits of the HPCI and RCIC systems to allow the systems to remain operable when equipment room cooling is unavailable.

Date of issuance: January 20, 1995.

Effective date: January 20, 1995.

Amendment Nos. 85 and 46.

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

Date of initial notice in **Federal Register**: September 14, 1994 (59 FR 47178).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 20, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 18,1994, as supplemented October 25, 1994.

Brief description of amendment: The amendment revises TS Section 3.14 (Fire Protection and Detection Systems—Limiting Conditions for Operation), TS Section 4.12 (Fire Protection and Detection Systems—Surveillances) and TS Section 6.0 (Administrative Controls) to relocate the fire protection requirements from the TSs to the IP3 Operational Specifications Manual. In addition, the amendment revised the IP3 Facility Operating License to include the NRC's standard fire protection license condition. These changes were made in accordance with the guidance provided in Generic Letter (GL) 86–10, ''Implementation of Fire Protection Requirements,'' and GL 88–12, ''Removal of Fire Protection Requirements from Technical Specifications.''

Date of issuance: January 13, 1995.

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

Amendment No.: 157.

Facility Operating License No. DPR–64: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: May 25, 1994 (59 FR 27065).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 13, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

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

Date of application for amendment: December 8, 1994.

Brief description of amendment: The amendment revises TS section 4.4.E.1 to allow a one-time extension to the 30-month interval requirement for leak rate testing of Residual Heat Removal containment isolation valves AC–732, AC–741, AC-MOV–743, AC-MOV–744 and AC-MOV–1870.

Date of issuance: January 13, 1995.

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

Amendment No.: 158.

Facility Operating License No. DPR–64: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 13, 1994 (59 FR 64223).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 13, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

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

Date of application for amendment: June 29, 1994, as supplemented December 2, 1994.

Brief description of amendment: The amendment revised Technical Specification (TS) Section 6.5, ''Review and Audit,'' and TS Section 6.8, ''Procedures,'' to establish a new review and approval process for nuclear safety-related procedures and to modify membership requirements for the Plant Operating Review Committee. The amendment also revised TS Section 6.5 to delete review and audit responsibilities for the Emergency and Security Plans consistent with Generic Letter 93–07, ''Modification of the Technical Specification Administrative Control Requirements for Emergency and Security Plans.''

Date of issuance: January 17, 1995.

Effective date: As of the date of issuance to be implemented within 90 days.

Amendment No.: 159.

Facility Operating License No. DPR–64: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 20, 1994 (59 FR 37081).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 17, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

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: June 17, 1994, as supplemented December 2, 1994.

Brief description of amendment: The amendment revises Section 6.5, ''Review and Audit,'' and Section 6.8, ''Procedures,'' of the Technical Specifications (TSs) to establish a new review and approval process for nuclear safety-related procedures. The amendment also revises Section 6.5 to modify membership requirements for the Plant Operating Review Committee and to delete review and audit responsibilities for the Emergency and Security Plans from the TSs consistent with Generic Letter 93–07, ''Modification of the Technical Specification Administrative Control Requirements for Emergency and Security Plans.''

Date of issuance: January 18, 1995.

Effective date: As of the date of issuance to be implemented within 90 days.

Amendment No.: 222.

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

Date of initial notice in **Federal Register**: July 20, 1994 (59 FR 37082).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 18, 1995.

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.

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

Date of application for amendment: September 19, 1994.

Brief description of amendment: The amendment revised the Technical Specifications for the snubber visual inspection schedule.

Date of issuance: January 20, 1995.

Effective date: January 20, 1995 and to be implemented within 90 days.

Amendment No. 68.

Facility Operating License No. NPF–58. This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 26, 1994 (59 FR 53843).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 20, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, Ohio 44081.

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 application for amendment: May 1, 1992 as clarified by facsimile transmission dated January 10, 1995.

Brief description of amendment: The amendment revises the LIMITING CONDITIONS FOR OPERATION and SURVEILLANCE REQUIREMENTS for the containment air locks, changes the exception for containment penetration status verification to include the annulus, clarifies containment air lock testing intervals, and clarifies the definition and bases for containment integrity.

Date of issuance: January 17, 1995.

Effective date: Date of issuance and to be implemented within 90 days.

Amendment No. 194.

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

Date of initial notice in **Federal Register**: September 2, 1992 (57 FR 40221).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 17, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606.

Washington Public Power Supply System, Docket No. 50–397, Nuclear Project No. 2, Benton County, Washington

Date of application for amendment: August 8, 1994.

Brief description of amendment: The amendment modifies the Technical Specifications (TS) to delete the requirement to obtain prior written relief from the Commission for inservice inspection (ISI) and inservice testing (IST) of components conducts pursuant to 10 CFR 50.55a. The amendment also adds a definition for the word ''biennial.''

Date of issuance: January 5, 1995.

Effective date: January 5, 1995.

Amendment No.: 133.

Facility Operating License No. NPF–21: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 14, 1994 (59 FR 56558).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 5, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Richland Public Library, 955 Northgate Street, Richland, Washington 99352.

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: November 23, 1993, as supplemented January 10, 12, and 13, 1995.

Brief description of amendments: These amendments revise the operating conditions and limiting conditions for operation for containment systems, and revise corresponding definitions and tests. In addition, the related bases are updated to ensure consistency and clarity.

Date of issuance: January 18, 1995.

Effective date: January 18, 1995, to be implemented within 45 days.

Amendment Nos.: 160 and 164.

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

Date of initial notice in **Federal Register**: January 19, 1994 (59 FR 2875).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 18, 1995.

The January 10, 12, and 13, 1995 submittals provided supplemental information that did not change the initial proposed no significant hazards consideration determination.

No significant hazards consideration comments received: No.

Local Public Document Room location: Joseph P. Mann Library, 1516 Sixteenth 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: October 27, 1993.

Brief description of amendment: The amendment changes Note 5 of Technical Specification Table 4.3–1 to reflect the use of integral bias curves, rather than detector plateau curves, to calibrate the source range instrumentation.

Date of issuance: January 9, 1995.

Effective date: January 9, 1995, to be implemented within 30 days.

Amendment No.: 83.

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

Date of initial notice in **Federal Register**: November 24, 1993 (58 FR 62159). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 9, 1995.

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: October 21, 1994, as supplemented by letters dated October 27, 1994 and December 2, 1994.

Brief description of amendment: This amendment revises Technical Specification (TS) Surveillance Requirements 4.7.1.2.1.c.2, operability testing of the auxiliary feedwater (AFW) pump auto start feature, and 4.3.2.2, engineered safety features (ESF) time response testing of the AFW pumps to exempt the testing of the turbine-driven AFW pump from the provisions of TS 4.0.4 for entry into Mode 3. In addition, TS Surveillance Requirement 4.7.1.2.1.c is revised to delete the requirement that the 18 month AFW surveillance be performed during shutdown.

Date of issuance: January 20, 1995.

Effective date: January 20, 1995, to be implemented within 30 days.

Amendment No.: 84.

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

Date of initial notice in **Federal Register**: November 23, 1994 (59 FR 60389) The December 2, 1994, supplemental letter provided clarifying information and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 20, 1995.

No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 25th day of January 1995.

For the Nuclear Regulatory Commission.

Jack W. Roe,

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

[FR Doc. 95–2350 Filed 1–31–95; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–35278; File No. SR–CBOE–95–02]

Self-Regulatory Organizations; Notice of Filing of Proposed Rule Change by the Chicago Board Options Exchange, Inc. Relating to the Listing of Long-Term Index Options Series (''LEAPS'') With a Duration of Up to Sixty Months Until Expiration

January 25, 1995.

Pursuant to Section 19(b)(1) of the Securities Exchange Act of 1934 (''ACT''),[1] and Rule 19b–4 thereunder,[2] notice is hereby given that on January 19, 1995, the Chicago Board Options Exchange (''CBOE'' or ''Exchange'') filed with the Securities and Exchange Commission (''Commission'') the proposed rule change as described in Items I, II, and III below, which Items have been prepared by the CBOE. The Commission is publishing this notice to

[1] 15 U.S.C. § 78s(b)(1) (1988).

[2] 17CFR 240.19b–4 (1991).