Westinghouse Electric Company proprietary information pursuant to 5 U.S.C. 552b(c)(4).

The agenda for the subject meeting shall be as follows:

Monday, May 11, 1998—8:30 a.m. until the conclusion of business.

Tuesday, May 12, 1998—8:30 a.m. until the conclusion of business.

The Subcommittee will continue its review of the results of the Westinghouse Test and Analysis Program supporting the AP600 design certification. Specifically, the Subcommittee will review issues pertaining to the AP600 Reactor Coolant System, including the resolution of issues identified in the February 19, 1998 ACRS letter. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the Westinghouse Electric Company, the NRC staff, their consultants, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the scheduling of sessions which are open to the public, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor, can be obtained by contacting the cognizant ACRS staff engineer, Mr. Paul A. Boehnert (telephone 301/415–8065) between 7:30 a.m. and 4:15 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Dated: April 16, 1998.

Sam Duraiswamy,

Chief, Nuclear Reactors Branch.

[FR Doc. 98–10663 Filed 4–21–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Subcommittee Meeting on Advanced Reactor Designs; Notice of Meeting

The ACRS Subcommittee on Advanced Reactor Designs will hold a meeting on May 13–15, 1998, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Wednesday, May 13, 1998—8:30 a.m. until the conclusion of business.

Thursday, May 14, 1998—8:30 a.m. until the conclusion of business.

Friday, May 15, 1998—8:30 a.m. until the conclusion of business.

The Subcommittee will continue its review of the Westinghouse AP600 design. Specifically, the Subcommittee will review Chapters 3, 6, 9A, 14, 16, and 17 of the AP600 Standard Safety Analysis Report, the probabilistic risk assessment (PRA), regulatory treatment of non-safety systems (RTNSS), and the associated NRC staff's draft Final Safety Evaluation Report. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, Westinghouse Electric Company, their consultants, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor, can be obtained by contacting the cognizant ACRS staff engineer, Mr. Noel F. Dudley (telephone 301/415–6888) between 7:30 a.m. and 4:15 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Dated: April 16, 1998.

Sam Duraiswamy,

Chief, Nuclear Reactors Branch.

[FR Doc. 98–10664 Filed 4–21–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to 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 March 30, 1998, through April 10, 1998. The last biweekly notice was published on April 8, 1998 (63 FR 17219).

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

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

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

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

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

By May 22, 1998, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's

Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

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

Date of amendment request: March 14, 1997.

Description of amendment request: The proposed amendment would delete license conditions which have been satisfied, revise others to delete parts which are no longer applicable or to revise references, and make editorial 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:

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

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

The license conditions were one-time commitments that have been satisfied. There are no physical changes to the facility, and all operating procedures, limiting conditions for operation, limiting safety system settings, and safety limits are unchanged. Removal of these license conditions is appropriate and safe.

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

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

Many of the proposed changes delete references to items that have been completed. The NRC required these items as a condition of granting the license. Since they have been satisfied as intended, deleting them is administrative.

None of the proposed changes affect the design or operation of any system, structure, or component in the plant. The safety functions of the related structures, systems, or components are not changed in any manner, nor is the reliability of any structure, system, or component reduced by the revised surveillance or testing requirements. The changes do not affect the manner by which the facility is operated and do not change any facility design feature, structure, system, or component. No new or different type of equipment will be installed. Since there is no change to the facility or operating procedures, and the safety functions and reliability of structures, systems or components are not affected, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The remaining changes are editorial in nature and have no impact on plant operation or design.

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

The proposed changes to the Operating License are generally administrative in nature and have no impact on the margin of safety of any Technical Specification. There is no impact on safety limits or limiting safety system settings. The changes do not affect any plant safety parameters or setpoints. The operating license conditions have been satisfied, as required. There are no changes to the conditions themselves. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Therefore, based on the above evaluation, Commonwealth Edison has concluded that these changes do not involve significant hazards considerations.

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

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

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

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: October 16, 1997.

Description of amendment request: The proposed amendment would add an exemption from 10 CFR 70.24(a) to the Unit 1 license consistent with the Unit 2 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:

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

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

There are no physical changes to the facility, and all operating procedures, limiting conditions for operation, limiting safety system settings, and safety limits are unchanged.

The specific requirements for granting an exemption from 10 CFR 70.24(a) have been met. The request is authorized by law, will not endanger life or property or the common defense and security, and is in the public interest.

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

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

The proposed change does not affect the design or operation of any system, structure, or component in the plant. The safety functions of structures, systems, or components are not changed in any manner, nor is the reliability of any structure, system, or component reduced by the revised surveillance or testing requirements. The change does not affect the manner by which the facility is operated and does not change any facility design feature, structure, system, or component. No new or different type of equipment will be installed. Since there is no change to the facility or operating procedures, and the safety functions and reliability of structures, systems, or components are not affected, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change to the Operating License has no impact on the margin of safety of any Technical Specification. There is not

impact on safety limits or limiting safety system settings. The change does not affect any plant safety parameters or setpoints. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Therefore, based on the above evaluation, Commonwealth Edison has concluded that the proposed change does not involve significant hazards considerations.

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

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

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

NRC Project Director: Stuart A. Richards.

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

Date of amendment request: March 13, 1998, as supplemented March 30, 1998.

Description of amendment request: The proposed amendment would revise the technical specifications (TS) to allow any two auxiliary feedwater (AFW) flow control valves to be inoperable concurrently for up to 72 hours, provided the corresponding redundant flow control valves and a pump in the other AFW train are operable.

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:

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

The proposed change would only alter the allowance for specific AFW flow control valves to be inoperable. It would not affect any operating limits, any plant operating conditions, or the physical capability of any plant equipment. Therefore, it would not affect the probability of any accident previously evaluated.

The proposed change would not reduce the AFW flow capability to the steam generators during operation under the affected Action Statement. It would allow more operational flexibility in plant operation when two AFW flow control valves in the same train were concurrently inoperable. The specified AOT [allowed outage time] of 72 hours would remain unchanged. Current TS allow continued operation for 72 hours with one of the three AFW pumps inoperable, or with one flow control valve in each train inoperable (provided the corresponding redundant flow control valve and a pump in the other pipe train are operable), but do not allow continued operation with both valves in the same train inoperable. The proposed change would allow any two valves to be inoperable, with the same provision that the corresponding redundant flow control valve and a pump in the other pipe train are operable.

Since, with the proposed change there would be no reduction in the ability to provide AFW flow to either steam generator, operation of the Facility in accordance with the proposed changes would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The changes do not alter the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The changes do allow different sets of AFW flow control valves to be inoperable, however, these changes retain a consistent level of AFW capability during operation under the Action Statement. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Therefore, operation of the Facility in accordance with the proposed TS change would not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change would not reduce the AFW flow capability to the steam generators during operation under the affected Action Statement. It would allow more operational flexibility in plant operation when two AFW flow control valves were concurrently inoperable. The specified AOT of 72 hours would remain unchanged.

Therefore, operation of the Facility in accordance with the proposed TS 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 Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Project Director: Cynthia A. Carpenter.

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

Date of amendment request: March 27, 1998 (NRC–98–0033).

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.5.2, ''ECCS—Shutdown,'' and TS 3.5.3, ''Suppression Chamber,'' raising the minimum water level required in the condensate storage tank (CST) to support the core spray system (CSS) when the suppression pool (the normal supply for CSS) is unavailable. The amendment would also eliminate incorrect information concerning CST inventory reserved for the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems. The associated Bases are also revised.

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 changes will not affect the performance or reliability of the Condensate Storage System which could lead to an accident because the Condensate Storage Tank (CST) is not involved as an initiator of any accident previously evaluated. The proposed change meets the design standards of the Condensate Storage System by providing assurance that sufficient water volume is available for the Core Spray System. This change also removes [an] erroneous discussion of water inventory for HPCI/RCIC Systems while in Operating [Operational] Conditions 4 and 5. The removal of information is acceptable since HPCI/RCIC Systems are not operable in these modes and will therefore not increase the probability of an accident. The increase in volume provides for vortex/air entrainment avoidance in the Core Spray System and will not increase consequences. Furthermore, the elimination of HPCI/RCIC information will not increase consequences of an accident previously evaluated because these systems are not credited for accident mitigation in Operating [Operational] Conditions 4 and 5.

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 add or modify any equipment or components related to the Condensate Storage System and will therefore not create any new failure modes or common failure modes. This proposed change raises the water level within the CST to ensure sufficient water volume is maintained and updates the TS by removing descriptive information with respect to CST water inventory for HPCI/RCIC Systems

while in Operating [Operational] Conditions 4 and 5. The Condensate Storage System will continue to operate as intended and as designed. This change will therefore not create the possibility of a new or different kind of accident.

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

The proposed change increases the required CST water level to provide at least 150,000 gallons of water available for the Core Spray System while maintaining adequate submergence of the Core Spray standpipe for avoiding vortex and air entrainment. As such, the proposed change involves no reduction on any margin of safety. Revision to TS Bases concerning discussion of reserve volume in CST for HPCI and RCIC, does not alter the requirement for Core Spray or Suppression Pool operability and does not involve a reduction in any margin of safety.

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

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

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

NRC Project Director: Cynthia A. Carpenter.

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

Date of amendment request: March 27, 1998 (NRC–98–0034).

Description of amendment request: The proposed amendment would clarify a footnote in Technical Specification (TS) 3.5.1, ''ECCS—Operating,'' and 3.5.2, ''ECCS—Shutdown,'' to indicate that a low pressure coolant injection system loop may be considered operable during alignment and operation for decay heat removal if it is capable of being manually realigned and is not otherwise inoperable. The associated Bases would also be revised.

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 changes involve actions required to realign the Low Pressure Coolant Injection (LPCI) system for LPCI injection if LPCI is required when operating in the Shutdown Cooling (SDC) mode. The additional actions described involve resetting isolations and trips which could occur prior to LPCI initiation. Resetting these logics does not initiate any valve operation or pump start; the LPCI initiation signals and interlocks remain in control of valve and pump logic.

The equipment interlocks that provide the isolation signal for the LPCI injection valves were designed to prevent drain down of the Reactor Pressure Vessel (RPV) when in SDC. The injection valve closure is the most conservative action in response to an RPV drain event. The current TS acknowledges that operator action to realign the suction path is necessary. The proposed change acknowledges that operator action to reset injection valve logic and pump trips is necessary. The time required to realign LPCI is not significantly different than the existing actions to realign the suction path.

No changes in either system design or operating strategies will be made as a result of these changes, thus no opportunity exists to increase the probability or consequences of a previously analyzed accident. Therefore, the proposed change does not involve a significant increase in the probability or consequences of a previously evaluated accident.

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

The manual realignment of the LPCI system from SDC following an isolation signal does not affect the accident analysis described in Chapter 15 of the UFSAR [updated final safety analysis report]. No new limiting single failure has been identified as a result of the proposed changes. The possibility of a new or different kind of accident from those previously analyzed will not be created by the change to the TS footnote or Bases, because the proposed change merely clarifies the actions necessary to realign the LPCI system. The time required to realign the system is not significantly different than the time necessary to realign the suction path. Therefore, no new or different types of failures or accident initiators are introduced by the proposed changes.

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

The proposed change described above affects the plant's ability to enter Operational Conditions 3, 4, and 5, and to achieve and maintain COLD SHUTDOWN conditions when shutting down the plant. The proposed change in combination with existing restrictions within the TS provide assurance that there is no credible mechanism to inhibit running the LPCI system. The minor additional operator action required to realign LPCI from SDC requires minimum time and effort considering controls for each division are located on their respective control panel. As a result of this change, there will be no changes in either system design or operating strategies because the proposed changes merely clarify existing TS requirements and actions necessary to meet TS requirements. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Cynthia A. Carpenter.

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

Date of amendment request: April 2, 1998 (NRC–98–0057).

Description of amendment request: The proposed amendment would permit entering Operational Conditions 1 and 2 prior to completion of Surveillance Requirements for the primary containment hydrogen and oxygen monitors in order to establish the conditions necessary (inerted containment) to properly perform the calibrations. The amendment would also increase the frequency of the calibration for the oxygen monitors from every 18 months to quarterly in accordance with vendor recommendations and correct the nomenclature for the hydrogen and oxygen monitors in tables 3.3.7.5–1 and 4.3.7.5–1.

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

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

The proposed change will permit delaying the performance of calibrations of the hydrogen and oxygen monitors until after the containment is inerted following a plant startup. The proposed change will also increase the calibration frequency for the oxygen monitors from once per 18 months to once per quarter, and change the nomenclature for the hydrogen and oxygen monitors.

The primary containment hydrogen and oxygen monitors are passive instruments that provide indication to control room operators of hydrogen and oxygen concentration in the primary containment. Because they perform only a passive monitoring function, the

hydrogen and oxygen monitors are not associated with the initiation of any previously evaluated accident. The indication provided by the monitors is used by the control room operators to ensure oxygen concentration remains below limits and to make decisions regarding the use of the Combustible Gas Control System, if necessary. The allowance to permit entry into applicable operational conditions before calibration ensures that the conditions (nitrogen environment) are appropriate for accurate calibration of the instrument. Delaying the calibration does not cause the instrument to cease to function. Calibrations verify and adjust, as necessary, the accuracy of the instrument to compensate for drift that may occur since the last calibration. Thus, even with a delayed calibration, the instruments still would provide valuable information to the operators. Consequently, this change will not involve a significant increase in the consequences of a previously evaluated accident because the monitors will still function and provide meaningful information until the calibration is completed.

The change to reduce the interval for calibration of the oxygen monitors from once per 18 months to once per quarter provides increased assurance of monitor accuracy and is consistent with the manufacturer's recommendations. Therefore, because this instrument is not associated with the initiation of an accident and the change improves the functionality of the instrument, the probability and consequences of previously evaluated accidents are not significantly affected.

The change in nomenclature is editorial, and, as such does not affect the probability or consequences of a previously evaluated accident.

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

As discussed above, the hydrogen and oxygen monitors are passive, indication-only instruments which provide information to control room operators. The proposed changes do not introduce a new mode of operation or involve a physical modification to the plant. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes involve the containment hydrogen and oxygen monitors which do not affect any parameters or assumptions used in the calculation of any safety margin with regard to Technical Specification Safety Limits, Limiting Safety System Settings, Limiting Control Settings or Limiting Conditions for Operation, or other previously defined margins for any structure, system, or component. 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: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161.

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

NRC Project Director: Cynthia A. Carpenter.

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

Date of amendment request: March 16, 1998.

Description of amendment request: The proposed amendment would revise technical specification (TS) Table 4.3–1 to add footnote 6 to the channel calibration requirement for all instrument channels that are provided with an input from neutron flux detectors. Footnote 6 provides that neutron detectors may be excluded from channel calibrations. Additional changes are proposed for BVPS–1 to provide consistency between BVPS–1 and BVPS–2. These additional changes would add channel calibration requirements to BVPS–1 TS Table 4.3–1 items 2.b. (Power Range, Neutron Flux, Low Setpoint), 5. (Intermediate Range, Neutron Flux), 6. (Source Range, Neutron Flux (Below P–10), and 23. (Reactor Trip System Interlocks P–6, P–8, P–9, and P–10). Furthermore, changes would be made to correct page numbers in the BVPS–2 Index and to add corresponding changes to the bases for both units.

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

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

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The protection systems will continue to function in a manner consistent with the plant design basis. The proposed changes will not affect any of the analysis assumptions for any of the accidents previously evaluated. The proposed changes will not affect the probability of any event initiators nor will the proposed changes affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the challenges imposed on safety-related equipment assumed to function during an accident. There will be no change to normal plant operating parameters or accident mitigation capabilities. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

There are no hardware changes associated with this license amendment nor are there any changes in the method by which any safety-related plant system performs its safety function. The normal manner of plant operation is unchanged.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of these changes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

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

The proposed changes do not affect the acceptance criteria for any analyzed event nor is there a change to any Safety Analysis Limit (SAL). Maintaining the SAL preserves the margin of safety.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. 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: B.F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001.

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: March 17, 1998.

Description of amendment request: The proposed amendment would revise Action 34 of technical specification (TS)

Table 3.3–3, ''Engineered Safety Feature Actuation System Instrumentation.'' Action 34 applies to Functional Units 6.b., ''Grid Degraded Voltage (4.16 kV Bus),'' and 6.c. ''Grid Degraded Voltage (480 v Bus).'' The proposed revision would require that with one channel inoperable, the inoperable channel be placed in the tripped condition within one hour; otherwise, the applicable action statement(s) for the associated emergency diesel generator made inoperable by the degraded voltage start instrumentation be entered immediately. The proposed revision would also require that with two channels inoperable, at least one of the two channels be restored to operable status and the other channel be placed in the tripped condition within one hour; otherwise, the associated emergency diesel generator shall be declared inoperable and its applicable action statement(s) shall be entered. In addition, corresponding changes would be made to the bases for TS 3/4.3.2 and the BVPS–2 Index would be revised to reflect changed page numbers.

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

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

Engineered Safety Feature Actuation System (ESFAS) will continue to function in a manner consistent with the plant design basis. The proposed change will not affect any of the analysis assumptions for any of the accidents previously evaluated. The proposed changes will not affect the probability of any event initiators. There will be no change to normal plant operating parameters. The emergency bus degraded voltage protection system is utilized for accident mitigation and is not considered to be the source of accidents previously evaluated.

Implementation of the proposed changes will now provide viable corrective actions which do not significantly increase the probability of failure of safety related equipment to perform its intended function. The proposed Action 34 permits a one hour time frame before the affected diesel generator(s) is required to be declared inoperable. This one hour period allows for repairs of most failures and takes into account the low probability of an event which would require the degraded voltage protection system to function. If adequate protection is not restored within this one hour period, the diesel generator(s) allowable outage time is invoked. The diesel generator(s) allowable outage time has been previously evaluated and determined to be an acceptable period of time during which plant operation may continue without an emergency backup power source. The loss of emergency bus degraded voltage protection is similar to the loss of the ability of an emergency diesel generator to provide electrical power to the safety related loads on the emergency buses. In both situations, a loss of offsite power, due to a total loss or a degraded condition, will result in the safety related loads not being capable of mitigating a design basis accident. The proposed changes to the Index page are administrative in nature and do not affect plant safety.

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

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

The emergency bus degraded voltage protection system is utilized for accident mitigation. The proposed changes will now provide viable corrective actions which do not result in a change in the manner in which the emergency bus loads are protected from a degraded voltage condition. These changes do not alter the function of the degraded voltage protection system. The proposed changes will continue to require that at least one of the two redundant 4160 volt or 480 volt emergency buses is protected from a degraded voltage condition assuming a single active failure of the opposite emergency bus degraded voltage protection system. This action will ensure that at least one train of engineered safety feature (ESF) equipment is not damaged due to a sustained bus undervoltage condition. The proposed addition of the requirement to enter the action statement for the inoperable diesel generator, if the one hour requirements of Action 34 cannot be met, will ensure that adequate compensatory actions to assure plant safety are taken. These requirements include the demonstration of the operability of the A.C. offsite sources by performing a specific surveillance within one hour and at least once per eight hours thereafter. If both diesel generators are inoperable, at least one diesel generator must be restored to operable status within two hours or the plant must be placed in cold shutdown within the following 36 hours.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of these changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of these changes.

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

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

The margin of safety is not significantly reduced because the A.C. electrical power sources will continue to provide sufficient capability, redundancy, and reliability to ensure availability of necessary power to ESF systems. The ESF systems will continue to function, as assumed in the safety analyses, to ensure that fuel, reactor coolant system and containment design limits are not exceeded. The proposed revisions to Action 34 will continue to require that at least one of the two redundant 4160 volt or 480 volt emergency buses is protected from a degraded voltage condition assuming a single active failure of the opposite emergency bus degraded voltage protection system. This action will ensure that at least one train of ESF equipment is not damaged due to a sustained bus undervoltage condition. The emergency loads, which are powered from that train of emergency buses, will continue to be available to perform their safety related functions. If the one hour requirements of Action 34 cannot be met, the affected emergency diesel generator will be declared inoperable. This will ensure that adequate compensatory actions to ensure plant safety are taken. The loss of emergency bus protection from a degraded voltage condition is similar to the loss of the ability of an emergency diesel generator to provide electrical power to the safety related loads on the emergency buses. In both situations, a loss of the offsite power, due to a total loss or a degraded condition, will result in the safety related loads not being capable of mitigating a design basis accident.

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

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

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

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: March 23, 1998.

Description of amendment request: The proposed amendment modifies Section 3.1.2 of the Technical Specifications (TS) to incorporate new pressure/temperature limits regarding reactor vessel pressurization heatup, cooldown, and inservice leak and hydrostatic leak test limitations in accordance with 10 CFR 50, Appendix G. These new limits would be applicable through the period of 17.7 effective full power years (EFPY) of operation.

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

consideration, which is presented below:

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated. The design basis event related to this change is nonductile failure of the reactor coolant pressure boundary. The updated pressure/temperature limits have been established in accordance with the requirements of 10 CFR 50, Appendix G. Revision of these curves for an applicability period of 17.7 EFPY is based on maintaining the required design margin. Operation of the facility in accordance with the proposed amendment provides assurance of protection against nonductile failure of the reactor coolant pressure boundary for operation through 17.7 EFPY. Therefore, operation in accordance with the proposed amendment does not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated. The design basis event related to the change is nonductile failure of the reactor coolant boundary. The proposed amendment provides assurance of protection against nonductile failure of the reactor coolant boundary for operation through 17.7 EFPY and is unrelated to the possibility of creating a new or different kind of accident.

3. Operation of the facility in accordance with the proposed amendment would not involve any reduction in a margin of safety since the design methodology has maintained the existing 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: Law/Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

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

NRC Project Director: Cecil O. Thomas.

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

Date of amendment request: March 27, 1997.

Description of amendment request: The proposed amendment, included as part of the proposed conversion from the current Technical Specifications (TS) to improved TS, would revise the Limiting Conditions for Operation in the event that one 250 V DC electrical power subsystem is inoperable.

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

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

The DC electrical power sources are used to support mitigation of the consequences of an accident; however, they are not considered the initiator of any previously analyzed accident. The proposed change merely provides direction to the operator to declare equipment associated with a 250 V DC electrical power subsystem inoperable if the subsystem becomes inoperable. This provides assurance that all affected features are immediately recognized as incapable of performing their safety functions, and requires immediate actions equivalent to those determined appropriate in the Technical Specifications for the affected features. Therefore, the proposed change does not involve an increase in the probability or consequences of any accident previously evaluated.

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

The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

This change does not involve a significant reduction in a margin of safety, since the proposed change results in establishing the level of safety for the loss of a 250 V DC electrical power subsystem equivalent to the level of safety that exists in the Technical Specifications for components and systems that are supplied by the 250 V DC electrical power system.

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

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

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

NRC Project Director: John N. Hannon.

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

Date of amendment request: March 2, 1998.

Description of amendment request: The proposed changes would revise the frequency for the performance of specific surveillances associated with the emergency diesel generators (EDGs) and delete the requirements contained in the current Technical Specifications for accelerated testing whenever the number of valid test failures associated with the EDGs is met or exceeded. In addition, the special requirements for reporting valid or invalid EDG failures would be deleted.

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 do not affect accident initiators or precursors and do not alter the design assumptions affecting the ability of the EDGs to mitigate the consequences of an accident.

Industry experience has indicated that excessive testing requirements have proven to be a contributor to increased EDG unavailability and equipment degradation. Removing inappropriate testing requirements increases EDG reliability and enhances the ability of EDGs to mitigate the consequences of an accident. Implementing the maintenance rule in accordance with 10 CFR 50.65, Regulatory Guide 1.160, and NUMARC 93–01 for the EDGs provides additional assurance that high EDG performance and availability will be maintained.

Deleting the special reporting requirements from the Technical Specifications is an administrative change that does not affect the ability of the EDGs to perform their specified safety function. North Atlantic will continue to notify the NRC of significant EDG failures in accordance with the provisions of 10 CFR 50.72 and 50.73.

Therefore, the proposed 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 previously analyzed.

The proposed changes do not alter the ability of the EDGs to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR). The proposed changes have no impact on component or system interactions, or the plant design basis. Instrumentation setpoints, starting,

sequencing and loading functions associated with the EDGs are not affected by the proposed changes. Furthermore, combining the implementation of the maintenance rule program with the proposed amendment will enhance both the availability and the performance of the EDGs.

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

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

There is no impact on equipment design or operation and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. The proposed changes do not affect the EDG's ability to ensure that sufficient power is available to supply the safety related equipment required for: 1) the safe shutdown of the facility, and 2) the mitigation and control of accident conditions within the facility. In addition, the proposed changes do not affect the EDG's ability to ensure that: 1) the facility can be maintained in a shutdown or refueling condition for extended periods of time, and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

EDG reliability and availability are expected to be improved by the proposed changes. Eliminating excessive testing requirements can improve safety by reducing challenges to plant systems and reducing equipment wear and degradation. While the proposed changes affect surveillance intervals there are no changes to the methods used to perform the surveillances. The surveillances will continue to demonstrate the ability of the EDGs to perform their intended function of providing electrical power to the emergency safety systems needed to mitigate design basis transients and accidents. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Project Director: Cecil O. Thomas.

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

Date of amendment request: March 5, 1998.

Description of amendment request: The proposed changes would revise the Seabrook Station Radiological Effluent Technical Specifications (TS) and Administrative Controls section of the Technical Specifications, as authorized by NRC Generic Letter (GL) 89–01, ''Implementation Of Programmatic Controls For Radiological Effluent Technical Specifications (RETS) In The Administrative Controls Section Of The Technical Specifications And The Relocation Of Procedural Details of RETS To The Offsite Dose Calculation Manual Or To The Process Control Program.'' The proposed amendment would incorporate programmatic controls in the TSs for radioactive effluents and for environmental monitoring conforming to the applicable regulatory requirements and would relocate the existing procedural details of the current RETS to the Offsite Dose Calculation Manual (ODCM). Procedural details associated with solid radioactive wastes would be relocated to the Process Control Program (PCP).

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 do not affect accident initiators or precursors and do not alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. The proposed changes do not alter or prevent the ability of structures, systems, or components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR). The proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details relative to radiological effluents.

Incorporation of programmatic controls for RETS in TSs will assure that the applicable regulatory requirements pertaining to the control of radioactive effluents will continue to be maintained. Since there are no changes to previous accident analyses, the radiological consequences associated with these analyses remain unchanged, therefore, the proposed 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 previously analyzed.

The proposed changes do not alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. The proposed changes have no impact on component or system interactions. The proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details relative to radiological effluents. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

There is no impact on equipment design or operation and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. The proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details relative to radiological effluents. A comparable level of administrative control will continue to be applied to those design conditions and associated surveillances being relocated to the ODCM or PCP. Therefore, the proposed changes do not involve a significant reduction in any margin of safety.

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

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Project Director: Cecil O. Thomas.

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

Date of amendment request: March 23, 1998.

Description of amendment request: The proposed change would revise the Seabrook Station Technical Specifications (TSs) to add a new TS 3.0.5 that would provide an exception to TSs 3.0.1 and 3.0.2 to allow the performance of required testing to demonstrate the operability of the equipment being returned to service or the operability of other equipment.

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 design basis accidents are not affected by the proposed administrative changes.

Specification 3.0.5 provides the administrative controls to ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed required testing. Specification 3.0.5 was incorporated in NUREG–1431, ''Standard Technical Specifications—Westinghouse Plants,'' (as modified by approved Technical Specification Task Force (TSTF) generic change Traveler TSTF–165), to address these, and other similar situations, that conflict with the requirements with the ACTIONS when equipment is returned to service. Specification 3.0.5 does not provide time to perform other preventative or corrective maintenance.

Inclusion of Specification 3.0.5 into the Seabrook Station Technical Specifications will provide operational flexibility with the restrictive compliance requirements of the other Applicability Specifications (3.0.1 and 3.0.2) and allow the performance of post-maintenance/surveillance activities to facilitate returning equipment to service or to allow other equipment to be tested. Therefore, inclusion of Specification 3.0.5 into the Seabrook Station Technical Specifications enhances plant safety by minimizing the potential for plant trip and/or transients. A qualitative risk assessment concerning returning components to service for post-maintenance testing was performed and concluded that the configurations allowed by Specification 3.0.5 have a negligible effect on the Seabrook Station risk profile. The components involved will have either completed calibration or maintenance, and can reasonably be expected to be able to perform their required safety function when returned to service for testing purposes. Therefore, the proposed 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 previously analyzed.

The proposed changes do not introduce new features or modify plant structures, systems and components or procedures that could possibly affect station operations under normal or abnormal conditions, thus, the potential for an unanalyzed accident is not created. The proposed administrative changes have no adverse affect on the safety limits or design basis accidents. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

There are no changes being made to the Technical Specification safety limits or safety system settings that would adversely affect plant safety. The changes do not affect the operation of structures, systems or components (SSCs) nor do they introduce administrative changes to plant procedures that could affect operator response during normal, abnormal or emergency situations. Inclusion of Specification 3.0.5 into the Seabrook Station Technical Specifications enhances plant safety by minimizing the potential for plant trip and/or transients by allowing equipment to be returned to service. A qualitative risk assessment concerning the return of components to service for post-maintenance testing was performed and concluded that the configurations allowed by Specification 3.0.5 have a negligible effect on the Seabrook Station risk profile. The components involved will have either completed calibration or maintenance, and can reasonably be expected to be able to perform their required safety function when returned to service for testing purposes. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Project Director: Cecil O. Thomas.

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

Date of amendment request: March 27, 1998.

Description of amendment request: The proposed change would revise Technical Specification (TS) 3.7.6, ''Control Room Emergency Makeup Air and Filtration (CREMAFS).'' The proposed change would modify the existing required action when both trains of CREMAFS are inoperable in Modes 5 and 6 by eliminating the restriction of suspending positive reactivity changes.

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

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

The proposed changes have no impact on the probability or consequences of an accident previously evaluated in the UFSAR. The control room ventilation systems are support systems which have a role in the detection and mitigation of accidents but do not contribute to the initiation of any accident previously evaluated. The removal of the positive reactivity addition restriction in Mode 5 and 6 has no impact on the course of any accidents previously evaluated. There are no presently evaluated positive reactivity or boron dilution accidents that credit the CREMAFS to mitigate its consequences or provide radiological protection. The positive reactivity restriction is overly restrictive in that it does not allow cooldown below 200° F when Mode 5 is entered as a result of both trains of CREMAFS being inoperable nor does it allow Reactor Coolant System temperature to vary.

The restriction is also redundant to Technical Specification 3.1.1.2 ''Reactivity Control Systems Shutdown Margin-T_{avg} less than or equal to 200° F'' in Mode 5 and Technical Specification 3.9.1 ''Refueling Operations Boron Concentration'' in Mode 6. Technical Specification 3.1.1.2 action, with shutdown margin less than the limit specified in the Core Operating Limits Report or with the Reactor Coolant System boron concentration less than 2000 ppm boron, requires immediate and continued boration until the restoration of the required shutdown margin or boron concentration. Similarly, Technical Specification 3.9.1 actions require suspension of core alterations or positive reactivity changes in addition to immediate and continued boration until the restoration of the required shutdown margin (K_{eff}) or boron concentration while in Mode 6. Sufficient shutdown margin ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with the postulated accident conditions are controllable within acceptable limits and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. The above referenced reactivity control system specifications provide the necessary protection for postulated reactivity addition accident conditions. Therefore, modifying the Technical Specification action that requires the suspension of positive reactivity changes and core alterations with both trains of the CREMAFS inoperable does not involve a significant increase in the 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 previously analyzed.

The proposed change that removes the positive reactivity addition restriction in Mode 5 and 6 does not create the possibility of a new accident nor does it create the possibility of a different kind of accident previously evaluated. There are no presently evaluated positive reactivity or boron dilution accidents that credit the CREMAFS to mitigate its consequences or provide radiological protection. The addition of positive reactivity during the above described situation is overly restrictive and furthermore redundant to Technical Specification 3.1.1.2 ''Reactivity Control Systems Shutdown Margin-T_{avg} less than or equal to 200° F'' in Mode 5 and Technical Specification 3.9.1 ''Refueling Operations Boron Concentration'' in Mode 6. The above referenced reactivity control system specifications provide the necessary protection for postulated reactivity addition accident conditions. Therefore, modifying the Technical Specification action that requires the suspension of positive

reactivity changes and core alterations with both trains of the CREMAFS inoperable does not create the possibility of a new or different accident from any previously evaluated.

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

The changes being proposed do not revise equipment design or operation nor do they make changes to Technical Specification required safety limits or safety system settings. In addition, they do not alter the environmental conditions which are to be maintained in the control room during normal operation and following an accident and they do not revise the accident analyses. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Project Director: Cecil O. Thomas.

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

Date of amendment request: April 3, 1998.

Description of amendment request: The proposed change would revise the Seabrook Station Technical Specifications (TSs) with administrative changes to support phased implementation of 24-month fuel cycle surveillance interval extensions. Specifically, the proposed change would: (1) provide wording changes in the Bases Section of TS 4.0.2 necessary to support 24-month surveillance interval extensions, (2) revise TS 4.0.5.b to provide revised terminology for inservice inspection and testing activities and their associated frequencies, (3) revise TS Table 1.1 to clarify current and future refueling intervals and their associated surveillance requirements and frequencies, and (4) delete the ''during shutdown'' restriction from the performance requirements of certain surveillance requirements.

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 design basis accidents are not affected by the proposed editorial and administrative changes. The proposed changes do not change the level of programmatic controls or the procedural details currently in place. Furthermore, these changes have no adverse affect to the safe operation of the station. Performance of certain maintenance and testing activities during conditions or modes other than shutdown will be evaluated by North Atlantic to ensure proper regard to their effect on safe operation of the plant is given prior to conduct of a particular surveillance, or portion thereof. Therefore, the proposed 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 previously analyzed.

The proposed changes do not introduce new features or modify plant structures, systems and components or procedures that could possibly affect station operations under normal or abnormal conditions, thus, the potential for an unanalyzed accident is not created. Performance of maintenance and testing activities on-line, as well as shutdown, are controlled by North Atlantic's procedures and policies to perform reviews and assessments of these activities to determine the affect on safe operation of the facility. The proposed editorial and administrative changes have no adverse affect on the safety limits or design basis accidents. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

There are no changes being made to the Technical Specification safety limits or safety system settings that would adversely affect plant safety. The changes do not affect the operation of structures, systems or components nor do they introduce administrative changes to plant procedures that could affect operator response during normal, abnormal or emergency situations. Performance of certain maintenance and testing activities during conditions or modes other than shutdown will be evaluated by North Atlantic to ensure proper regard to their effect on safe operation of the plant is given prior to conduct of a particular surveillance, or portion thereof. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Project Director: Cecil O. Thomas.

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

Date of amendment request: April 1, 1998.

Description of amendment request: The proposed revision to the Millstone Unit 3 licensing basis would add a new sump pump subsystem to address groundwater inleakage through the containment basemat.

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

NNECO has reviewed the proposed revision in accordance with 10 CFR 50.92 and has concluded that the revision does not involve a significant hazards consideration (SHC). The bases for this conclusion is that the three criteria of 10 CFR 50.92(c) are not satisfied. The proposed revision does not involve an SHC because the revision would not:

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

The current FSAR [Final Safety Analysis Report] credits the waterproof membrane for assuring that groundwater inleakage is not significant and would have no impact on safety related structures and components. However, degradation of the waterproof membrane has been detected, and it is now concluded that groundwater inleakage can be significant in that it could affect the operability of the RSS [recirculation spray system] pumps. The original plant design had only nonsafety-related RSS sump pumps available for pumping the groundwater from the RSS sumps. These pumps are not powered from the emergency busses and would not be accessible during a design basis LOCA [loss-of-coolant accident].

Thus, it is assumed that they would not be available to mitigate a design basis accident. Two independent safety-related air-driven sump pumps have been installed to eliminate the potential for groundwater inleakage that would affect the RSS pumps.

Air-driven sump pumps have been installed with the air supply line routed to a connection outside the ESF [engineered safety features] building. This allows the installation of an air compressor in an area that is accessible during a design basis accident such as a LOCA. Two air compressors have been staged in designated locations, and will be maintained and

periodically tested to ensure their availability. Periodic testing of the sump pumps will also be performed. The surveillance requirements have been incorporated into the Technical Requirements Manual.

EOP [Emergency Operating Procedure] 35–ES1.3 has been modified to add a step to install the compressors and start the sump pumps. It is estimated that these sump pumps would be needed approximately ten hours after a design basis accident. Thus, there is sufficient time for the operators to perform this action. Since sufficient time is available, the action has been incorporated into procedures and the environmental conditions allow access to the area, it is concluded that credit for operator action can be taken.

Thus, the new system is single failure proof and meets the requirements of Standard Review Plan 3.4.1 which stated the following:

''If safety-related structures are protected from below-grade groundwater seepage by means of a permanent dewatering system, then the system should be designed as a safety-related system and meet the single failure proof criterion.''

This provides assurance that the RSS pumps and other safety-related structures and components will perform the required safety function as assumed in the accident analysis.

The current nonsafety-related RSS sump pump system will continue to provide protection from groundwater inleakage during normal operation. Thus, there is no impact on the probability of occurrence of a transient because of equipment or structural failure due to groundwater inleakage. In addition, the new safety-related RSS sump pump system provides additional assurance that groundwater inleakage would not affect structures or equipment during an extended loss of offsite power or a design basis accident. Thus, it is concluded that there is no impact on the probability of occurrence of any previously evaluated accident.

The change results in the use of the new air-driven sump pumps to remove groundwater in-leakage from the RSS cubicles. To preclude the possibility for radiological contamination of the groundwater, all sources of liquid radiological contamination to the sumps have been eliminated. The RSS cubicle floor drains leading to Sumps 7A/7B have been plugged. Drains from equipment determined *not* to be a potential source of radiological contamination continue to drain to Sumps 7A/7B (sources include CCP [component cooling water] and Service Water relief valves) and are covered with splash guards to prevent the entrance of contaminated spray. The Hydrogen Recombiner area floor drains and the drain from the PASS [post accident sampling system] sample sink, all of which are nonsafety-related, have been isolated from the indirect waste receptor which drains to Sump 7B. Sumps 7A and 7B have been cleaned and the existing nonsafety-related sump pumps replaced to remove any existing residual contamination. The nonsafety-related pumps (3DAS–P8A/B) discharge to ESF Building sump 3DAS–SUMP 10. To preclude any potential siphoning from the potentially contaminated Sump 10 back to Sumps 7A/7B, the lines of the existing nonsafety-related pumps have been shortened to discharge above the water level in Sump 10.

The walls of Sumps 7A/7B have been extended to protect from a Limited Passive Failure and Pipe Break in the RSS cubicles. The expected flooding height is 6.6 inches [ ]. The sump cubicle height was extended to 3 ft. above the cubicle floor, well above this height. The sumps are covered with a vented hood to protect from pipe break spray and miscellaneous overhead leaks to further assure the sumps remain isolated from potentially contaminated RSS system fluids.

The existing SLCRS [supplementary leak collection and release system] boundary has been extended to the isolation valves located outside of the ESF building. Additionally, when the sump level is reduced while using the air driven pump, the pumps are designed to prevent air from being discharged through the pump discharge outside of the ESF building.

Thus, use of the new sump pumps would not affect the offsite doses following a design basis accident.

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

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

The current nonsafety-related RSS sump pump system will continue to provide protection from groundwater inleakage during normal operation. This will continue to provide assurance there is no potential for a transient because of equipment or structural failure due to groundwater inleakage. In addition, the new safety-related RSS sump pump system provides additional assurance that groundwater inleakage would not affect structures or equipment during an extended loss of offsite power or a design basis accident.

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

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

The current FSAR credits the waterproof membrane for assuring that groundwater inleakage is not significant and would have no impact on safety related structures and components. However, degradation of the waterproof membrane has been detected and it is now concluded that groundwater inleakage can be significant in that it could affect the operability of the RSS pumps. Original design had only nonsafety-related RSS sump pumps available for pumping the groundwater from the RSS sumps. These pumps are not powered from the emergency busses and would not be accessible during a design basis LOCA. Thus, it is assumed that they would not be available to mitigate a design basis accident. Two independent safety-related air-driven sump pumps have been installed to eliminate the potential for groundwater inleakage that would affect the RSS pumps. The new system is single failure proof and meets the requirements of Standard Review Plan 3.4.1.

Use of the new system requires operator action to install pre-staged air compressors to provide power for the new air-driven sump pumps. It is estimated that these sump pumps would be needed approximately ten hours after a design basis accident. Thus, there is sufficient time for the operators to perform this action. Since sufficient time is available, the action has been incorporated into procedures and the environmental conditions allow access to the area, it is concluded that credit for operator action can be taken.

With credit for the new single failure proof air-driven sump pumps and operator action to install pre-staged compressors to provide power for the pumps, the new subsystem provides the required assurance that the RSS pumps will not be affected by groundwater inleakage. Thus, it is concluded that the RSS pumps would be operable for long term accident mitigation and there is no impact on the margin of safety as defined in the basis of the Emergency Core Cooling Technical Specifications or any other Technical Specification.

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

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

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

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

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

NRC Deputy Director: Phillip F. McKee.

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

Date of amendment request: December 23, 1997.

Description of amendment request: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant, Unit Nos. 1 and 2 to revise Technical Specification (TS) 3/

4.7.11, Table 3.7–1, ''Maximum Allowable Power Range Neutron Flux High Setpoint With Inoperable Steam Line Safety Valves.'' The power range (PR) neutron flux high setpoints would be changed based on revised calculational methodologies for 1, 2, or 3 inoperable MSSVs per steam generator (SG). The proposed TS change would lower the PR neutron flux high setpoints when 2 or 3 MSSVs are inoperable per loop such that the maximum power level allowed would be within the heat removing capability of the remaining operable MSSVs. Although the method for calculating the maximum power level allowed when one MSSV per loop is inoperable has been revised, the results have not and the limit remains the same. The associated Bases would also be revised.

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 reduction in the power range (PR) neutron flux high setpoint Technical Specification (TS) values does not initiate an accident. Technician adjustments to lower the PR neutron flux high setpoints could cause a reactor trip (RT). However, this action is already a TS requirement. Thus, reducing the TS setpoint values from their current values will not change the requirement for a technician to adjust the setpoints downward when main steam safety valves (MSSVs) become inoperable, and therefore, will not increase the probability of a RT.

The reduction of the setpoints assures that the consequences of an accident when the MSSVs are inoperable are not affected by assuring that the MSSVs will continue to prevent overpressure of the main steam leads and steam generators (SGs) and remove adequate heat for the reactor coolant system.

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

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

Reduction of the PR neutron flux high setpoints does not change the method by which any safety-related system performs the function.

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

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

With the reduction in the PR neutron flux high setpoints for inoperable MSSVs, the MSSVs will still prevent SG pressure from exceeding 110 percent of SG design pressure in accordance with the ASME code. The change is conservative. The conclusions for the Final Safety Analysis Report Update accident analyses are unaffected by the change, remain valid, and provide margin.

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

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

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

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

NRC Project Director: William H. Bateman.

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

Date of amendment request: February 6, 1998.

Description of amendment request: The proposed changes would revise the Reactor Protection System (RPS) Normal Supply Electrical Protection Assembly (EPA) Undervoltage Trip setpoint to reflect a reanalysis of the most limiting applied load minimum voltage requirements.

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

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

The proposed setpoint change evaluated in Section III does not involve any physical changes to the plant, does not alter the way these systems function, and will not degrade the performance of the plant safety systems. The proposed instrument setpoint changes ensures that plant safety limits are not exceeded for the most limiting voltage requirements. The type of testing and the corrective actions required if the subject surveillances fail remains the same. The proposed changes do not adversely affect the reliability of these systems or affect the ability of the systems to meet their design objectives. A historical review of surveillance test results supports these conclusions.

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

The proposed setpoint change evaluated in Section III does not modify the design or operation of the plant, therefore, no new failure modes are introduced. The proposed instrument setpoint change ensures that plant safety limits are not exceeded for the most limiting voltage requirements. No changes are proposed to the type and method of testing performed. A historical review of surveillance test results supports these conclusions.

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

The proposed setpoint change evaluated in Section III results in minimal impact on system reliability in the interval between surveillance tests. This is based on the redundant design of the evaluated systems. A review of past surveillance history has shown no evidence of failures which would significantly impact the reliability of these systems. Operation of the plant remains unchanged by this proposed setpoint change. The assumptions in the Plant Licensing Basis are not adversely impacted. Therefore, the proposed changes do not result in 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: S. Singh Bajwa, Director.

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

Date of amendment request: February 6, 1998.

Description of amendment request: The proposed changes would allow reactor coolant system pressure tests to be conducted in Cold Shutdown Mode. Primary containment integrity is not required in this mode, facilitating containment access for inspections. The proposed changes also allow some outage activities on other systems to continue during the pressure testing. The licensee claims the proposed changes are consistent with the Boiling Water Reactor Standard Technical Specifications given in NUREG–1433, Revision 1.

Basis for proposed no significant hazards consideration determination:

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

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

The probability of a leak in the reactor coolant pressure boundary during reactor coolant system pressure testing is not increased by considering the reactor to be in Cold Shutdown. Since the pressure tests are performed nearly water solid, at low decay heat values, and near Cold Shutdown conditions, the stored energy in the reactor core will be low. Under these conditions, the potential for failed fuel and a subsequent increase in coolant activity is minimized. In addition, Special Operations LCO [Limiting Condition for Operation] 3.12.A requires supporting LCOs for ECCS [emergency core cooling system]-Cold Condition, Standby Gas Treatment, Secondary Containment isolation and Standby Gas Treatment initiation instrumentation, and Auxiliary Electrical Systems to be met to ensure secondary containment integrity is maintained and capable of handling any airborne radioactivity or steam leaks that could occur during the performance of hydrostatic or leak testing. A listing of secondary containment isolation valves required to maintain Secondary Containment Integrity is included in plant controlled procedures. The required pressure testing conditions provide adequate assurance that the consequences of a steam leak will be conservatively bounded by the consequences of the postulated main steam line break outside of primary containment. In the event of a large primary system leak, the reactor vessel would rapidly depressurize, allowing the low pressure core cooling systems to operate. The capability of these systems would be adequate to keep the core flooded under this low decay heat load condition. Small system leaks would be detected by leakage inspections before significant inventory loss occurred. Therefore, the consequences of an accident previously evaluated are not significantly increased.

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

The proposed changes do not introduce any new accident initiators or failure mechanisms since the changes do not involve any changes to structures, systems, or components, do not involve any change to the operation of systems, and alter procedures only to the extent that the 212° F limit may be exceeded during reactor coolant system pressure testing with certain systems inoperable. There are no alterations to plant systems designed to mitigate the consequences of accidents. The only difference is that a different subset of plant systems would be utilized for accident mitigation than those utilized during the Hot Shutdown Mode. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from those previously evaluated.

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

Since pressure tests are performed nearly water solid, at low decay heat values, and near Cold Shutdown conditions, the stored energy in the reactor core will be low. Under these conditions, the potential for failed fuel and a subsequent increase in coolant activity is minimized. Since secondary containment integrity will be maintained, in accordance with the Special Operations LCO, the secondary containment will be capable of handling any airborne radioactivity or steam leaks that could occur during the performance of hydrostatic or leak testing. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: S. Singh Bajwa, Director.

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

Date of amendment request: February 26, 1998.

Description of amendment request: The proposed changes would change the allowed containment leakage rate to 1.5 percent per day, changes the assumed standby gas treatment system (SBGT) filter efficiency, and revises reactor coolant sampling requirements for low Iodine-131 concentrations.

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 Authority [the licensee] has evaluated the proposed TS [technical specification] Amendment and determined that it does not represent a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, operation of the James A. FitzPatrick Nuclear Power Plant in accordance with the proposed amendment will not:

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

The proposed changes do not involve a change to the design or operation of the plant. The systems affected by this proposed TS change are not assumed in any safety analyses to initiate any accident sequence. Therefore, the probability of any accident previously evaluated is not increased by this proposed TS change. The change in the allowable containment leakage rate (L_a) is consistent with the accident analyses. The assumption of only 90% SBGT filter efficiency is conservative with actual system performance and is consistent with Regulatory Guide 1.52. There is no significant change to the consequences of an accident previously evaluated because maintaining containment leakage within limits assumed in the accident analyses ensures that the dose consequences resulting from an accident are not increased. The calculated doses with the decreased SBGT system charcoal efficiency for design basis accidents are marginally increased but still meet, and are well below, the dose acceptance criteria of 10 CFR 100, the SRP [Standard Review Plan, NUREG–0800], and GDC [General Design Criterion] 19 of Appendix A to 10 CFR 50. The proposed TS changes maintain an equivalent level of reliability and availability for all affected systems. The ability of the affected systems associated with maintaining leak rate integrity to perform their intended function is unaffected by the proposed TS changes. Implementation of these changes will provide continued assurance that specified parameters associated with containment integrity will remain within acceptance limits, and as such, will not significantly increase the consequences of a previously evaluated accident. The change in the value of .007 [microcurie]/ml to .002 [microcurie]/ml in section of 4.6.C. ''Coolant Chemistry'' is a minor editorial change, is more conservative, and will correct the inconsistency between the technical specification and its basis and as such, will not significantly increase the consequences of a previously evaluated accident.

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

The proposed amendment changes the allowed containment leakage rate to 1.5%, changes the assumed value for SBGT system charcoal filter efficiency, and changes a specification in section of 4.6.C. ''Coolant Chemistry'' from the value of .007 [microcurie]/ml to .002 [microcurie]/ml. No new accident modes are created by clarifying the numerical value of the allowable containment leakage rate (L_a) or changing the assumed value for the SBGT system charcoal filter efficiency. No safety-related equipment or safety functions are altered, or adversely affected, as a result of these changes. The proposed changes will not introduce failure mechanisms beyond those already considered in the current plant safety analyses. Changing the allowable leakage rate, the assumed value for the efficiency of the SBGT system charcoal filter, and the specification in the bases section of 4.6.C. ''Coolant Chemistry'' does not contribute to the possibility of a new or different kind of accident or malfunction from those previously analyzed.

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

The proposed amendment changes the allowed containment leakage rate to 1.5%, changes the assumed value for SBGT system charcoal filter efficiency, and changes a specification in section of 4.6.C. ''Coolant Chemistry'' from the value of .007 [microcurie]/ml to .002 [microcurie]/ml. The design of the FitzPatrick plant is not changed. The methodology for test performance is unchanged and Type A, B and C tests will continue to be performed at [greater than or equal to] P_a. The value of L_a specified in proposed specification 6.20 is consistent with the accident analyses, therefore, the dose consequences of any analyzed accidents are not increased as a result of this change. The calculated doses as a result of the decrease in the assumed efficiency of the SBGT system charcoal filters for design basis accidents are marginally increased but still meet, and are well below, the dose acceptance criteria of 10 CFR 100, the SRP, and GDC 19 of Appendix A to 10 CFR 50. The change in the specification in section 4.6.C. ''Coolant Chemistry'' from .007 [microcurie]/ml to .002 [microcurie]/ml is a minor editorial change, is more conservative, and will correct the inconsistency between the technical specification and its basis. Therefore, the proposed changes provide continued assurance of the leak tightness of the containment and conservatively assume SBGT system charcoal filter efficiency for the purpose of dose calculations for design basis accidents without adversely affecting the public health and safety and, as such, will not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: S. Singh Bajwa, Director.

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

Date of amendment request: March 30, 1998.

Description of amendment request: The proposed changes would change the interval of selected Logic System Functional Tests (LSFT) from semiannually to once per 24 months. The definition of LSFT is also revised to be consistent with the Boiling Water Reactor 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. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The only significant change proposed by this application involves the extension of the surveillance test interval for the LSFTs required by the TS. The other changes involve editorial, format, and clarification changes, which by their nature are of no safety significance.

Extending the LSFT interval from semiannually to once per 24 months does not involve plant physical changes, change any TS setpoints, or introduce any new mode of plant operation. Therefore, the change does not degrade the performance of any safety system assumed to function in the accident analysis, and therefore, will not increase the consequences of an accident.

Extending the LSFT interval from semiannually to 24 months results in no significant change in the logic system unavailability due to equipment failure. The reliability of safety systems subject to the LSFT are dominated by that of the mechanical components, and the logic system circuit relay coils which are subject to the more frequent functional test requirements. These factors are confirmed by the availability record of the affected safety system based on the past surveillance test history. Furthermore, the longer test intervals reduce the unavailability due to testing for the applicable safety system while the plant is operating. For these reasons, there is not a significant increase in the probability of an accident.

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

The proposed changes do not introduce any new accident initiators or failure mechanisms since the changes do not introduce any new modes of plant operation, make any physical changes, or change any TS setpoints. The changes reduce the probability of accidents initiated by test-induced plant transients by reducing the number of times the tests must be performed.

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

The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. In several aspects, the proposed changes may actually enhance the margin of safety by reducing the potential for test-induced plant transients, reducing the unavailability due to test of the applicable safety system, and reducing any potential incremental logic system component wear. For these reasons, the 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: S. Singh Bajwa, Director.

Public Service Electric & Gas Company, Docket No. 50–272, Salem Nuclear Generating Station, Unit No. 1, Salem County, New Jersey

Date of amendment request: March 26, 1998.

Description of amendment request: The proposed amendment would revise Technical Specification 3.1.3.3, ''Rod Drop Time,'' to change the applicability from Mode 3 (hot shutdown) to Modes 1 and 2 (startup and 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. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to the Technical Specification Mode applicability provides consistency between the testing requirements as stated in the surveillance requirement of the Technical Specifications and intended by the initial conditions specified in the limiting condition for operations. The proposed change does not introduce any physical changes to the plant or equipment already in place in the plant, the proposed change ensures that testing of the rod drop times is performed in a manner that is consistent with the Technical Specifications and the assumptions made in the Salem accident analysis.

Therefore, the proposed amendment does not increase the probability or consequences of any accident previously evaluated.

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

The proposed change does not introduce a new component or changes the manner in which the facility is operated, maintained or tested. Thus no new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed change to the facility.

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

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

As stated in question number 2, the proposed change does not introduce a new component or changes the manner in which the facility is operated. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety. The Technical Specifications remain the same, as the input, or initial conditions, of the safety analysis have not changed. Therefore, there is no reduction in the margin to 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, NJ 08079.

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Project Director: Robert A. Capra.

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

Date of amendment request: March 3, 1998.

Description of amendment request: The proposed amendment would change the Browns Ferry Nuclear Plant Unit 2 and Unit 3 Technical Specification Figure 3.6–1 which contains the reactor vessel pressure-temperature (PT) limits. The change would extend the validity of the curves to 32 effective full-power years (EFPY). The current PT curves are effective up to 12 EFPY. In addition to revised PT curves, several changes to the notes applicable to the curves are also proposed to be consistent with the supporting analysis.

The proposed PT curves also would support a planned 5% power increase for each unit. Approval of the proposed power increase is pending and is the subject of a separate action before the Commission.

The Tennessee Valley Authority has submitted the proposed change in current technical specification (CTS) format and in the improved standard technical specification (ISTS) format. Conversion to the ISTS format is pending.

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

A. The changes do not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)) because the proposed changes merely adjust the reference temperature for the limiting reactor vessel beltline material to account for accumulated and projected irradiation effects. The adjusted reference temperature analyses were performed in accordance with the requirements of Appendix G of 10 CFR part 50 and the guidance contained in Regulatory Guide 1.99, Revision 2. The changes do not otherwise affect the manner by which the facility is operated and do not change any facility design feature or equipment. Since the protection previously provided will continue to be provided and there is no change to the facility or operating procedures, there is no effect upon the probability or consequences of any accident previously analyzed.

B. The changes do not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because no new failure modes are introduced. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because they do not affect the function of any facility structure, system or component, or affect the manner by which the facility is operated.

C. The changes do not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)) because the proposed changes assure that the reactor vessel PT limits will be valid for operation up to 32 EFPY and that the safety margins specified in Appendix G of 10 CFR part 50 will be maintained.

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: Athens Public Library, 405 E. South Street, Athens, Alabama 35611.

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

NRC Project Director: Frederick J. Hebdon.

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: February 13, 1998 (TS 97–03).

Brief description of amendments: The proposed amendments would change the Sequoyah (SQN) Technical Specification (TS) by adding a new limiting condition for operation (LCO) that addresses requirements for the main feedwater isolation, regulating, and bypass valves.

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

TVA has concluded that operation of SQN Units 1 and 2 in accordance with the proposed change to the TSs [or operating license(s)], does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

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

TVA will not change plant components, functions, or operating practices by implementing a change that adds a TS requirement for the main feedwater isolation, regulating, and bypass valves. TVA will maintain and verify operability of these valves through the proposed surveillance and actions to ensure the accident mitigation functions are available when applicable. These valves are not considered to be the source of an accident and the conservative addition of a requirement to maintain their safety function will not increase the probability of an accident. TVA will not increase the consequences of an accident by implementing this change because this addition ensures that the isolation of main feedwater is available to mitigate the consequences of an accident.

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

TVA will not alter plant equipment or operating activities in the implementation of the proposed TS change. The valves used for the isolation of main feedwater are not a potential source for accidents and are designed for accident mitigation purposes. Therefore, TVA will not create the possibility of an accident of a different kind.

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

TVA maintains and ensures the availability of the isolation function for the main feedwater system as assumed in the SQN accident analysis. TVA proposes this TS change to further assure this capability and to meet the requirements of 10 CFR 50.36. TVA will not change the methods of operating the plant or setpoints associated with safety-related equipment in the implementation of this request. Therefore, TVA will not reduce the margin of safety by implementing a TS LCO for the isolation functions of the main feedwater system.

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

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

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

NRC Project Director: Frederick J. Hebdon.

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: February 13, 1998 (TS 97–07).

Brief description of amendments: The proposed amendments would change the Sequoyah (SQN) Technical Specification (TS) requirements for main steam isolation valves (MSIVs) to incorporate MSIV requirements consistent with the Westinghouse Standard TS (NUREG–1431) and would add testing requirements for the MSIVs that ensure the valves close on an actual or simulated automatic actuation signal.

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

TVA has concluded that operation of SQN Units 1 and 2, in accordance with the proposed change to the TSs, does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

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

The proposed changes provide enhancements and clarifications of the requirements for inoperable MSIVs and periodic testing provisions. These changes do not alter the safety functions of the MSIVs or the operating practices that govern their application to plant conditions. The actions for Modes 2 and 3 are revised such that a longer time could occur before an inoperable MSIV is closed or the unit is placed in a mode that does not apply. However, this increase will not significantly impact the ability of the valves to mitigate an accident or affect the accident generation possibility. This is based on the low probability of an accident occurring that would require closure of the MSIVs and reasonable time intervals to transition to lower modes based on operating experience to reach the required modes in an orderly manner without challenging unit systems.

The MSIVs provide accident mitigation functions but do not contribute to accident generation. The MSIV functions have not been altered by the proposed changes. Therefore, the proposed changes will not increase the probability of a previously evaluated accident. Based on the above discussions, the proposed changes will not significantly increase the consequences of an accident and in some instances they will enhance the safety functions.

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

The primary function of the MSIVs is to support accident mitigation and are not a significant contributor to events that could generate accidents. The main impact that could result from an inoperable MSIV is an inadvertent closure that results in a unit trip. This event is bounded by the accidents that are currently evaluated for SQN. Since the proposed change does not alter MSIV functions and the new surveillance will be performed in modes that will not challenge unit systems, the possibility of a new or different kind of accident is not created.

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

The proposed changes clarify and enhance the current SQN requirements for the MSIVs with one exception. This exception is the completion time added to the Modes 2 and 3 action that could be a negative impact to the margin of safety. This change could allow the MSIV safety function to be inoperable for a longer period of time. The overall effect of the proposed changes considering the additional end-device testing, periodic verification of inoperable MSIV closure, and removal of the action to allow MSIV closure in Mode 1, is considered a positive impact to the margin of safety. Therefore, there is not a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Frederick J. Hebdon.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: March 25, 1998.

Brief description of amendments: The amendments revise the Technical Specifications (TS) Sections 6.1.1; 6.2.1.b; 6.5.1.1; 6.5.1.6. a, d, h, and m; 6.5.1.7.c; 6.5.1.8; 6.14.1.2; 6.15.b; 6.2.3.5; 6.5.1.2; 6.5.1.7.a for Unit 1 and 6.1.1; 6.2.1.b; 6.5.1.1; 6.5.1.6. a, d, h, and m; 6.5.1.7.c; 6.5.1.8; 6.13.b; 6.14.b; 6.2.3.5; 6.5.1.2; and 6.5.1.7.a for Unit 2, changing the title of Station Manager to Site Vice President, and the titles of the Assistant Station Managers to Manager-Station Operation and Maintenance and Manager-Station Safety and Licensing.

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:

Virginia Electric and Power Company has reviewed the proposed Technical Specifications changes against the criteria of 10 CFR 50.92 and has concluded that the changes do not pose a significant hazards consideration. Specifically, station operations in accordance with the proposed Technical Specifications changes will not:

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

The proposed changes are administrative in nature. The overall responsibility for safe operation and review of plant operations is not being changed. There are no changes to the operation of any plant system or its design as a result of these changes. Therefore, neither the probability of occurrence nor the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report are increased.

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

The proposed changes are administrative in nature. The overall responsibility for safe operation and review of plant operations is not being changed. There are no changes to the operation of any plant system or its design that could create any new modes of operation or accident precursors. Therefore, it is concluded that no new or different kind of accident or malfunction from any previously evaluated has been created.

c. The proposed changes do not result in a significant reduction in margin of safety as defined in the basis for any Technical Specifications.

The proposed changes are administrative in nature. The overall responsibility for safe operation and review is not being changed. There are no changes to the operation of any plant system or its design as a result of these changes. Safety systems are maintained operable as required by Technical Specifications. Therefore, the margin of safety is not changed.

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: The Alderman Library, Special

Collections Department, University of Virginia, Charlottesville, Virginia 22903–2498.

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Project Director: P. T. Kuo, Acting Project Director.

Virginia Electric and Power Company, Docket Nos. 50–280 and 50–281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia

Date of amendment request: November 5, 1997.

Description of amendment request: The proposed Operating License change and changes to the technical specifications (TS) would permit the use of a temporary alternate supply line (jumper) to provide service water (SW) to the component cooling heat exchangers. The temporary jumper will permit maintenance to be performed on the existing supply line.

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:

Virginia Electric and Power Company has reviewed the proposed changes against the criteria of 10 CFR 50.92 and has concluded that the changes do not pose a significant safety hazards consideration as defined therein. The proposed Operating License and Technical Specifications and Bases changes are necessary to allow the use of a temporary, seismic, non-missile protected jumper to provide service water (SW) to the Component Cooling Heat Exchangers (CCHXs) while maintenance work is performed on the existing SW supply line to the CCHXs. Since there is only one SW supply line to the CCHXs, an alternate SW supply must be provided whenever the line is removed from service. The temporary jumper provides this function.

The use of the temporary jumper has been thoroughly evaluated, and appropriate constraints and compensatory measures (including a Contingency Action Plan) have been developed to ensure that the temporary jumper is reliable, safe, and suitable for its intended purpose. A complete and immediate loss of SW supply to the operating CCHXs is not considered credible, given the project constraints and the unlikely probability of a generated missile. Existing station abnormal procedures already address a loss of component cooling, and the use of alternate cooling for a loss of decay heat removal, in the unlikely event that they are required. Furthermore, appropriate mitigative measures have been identified to address potential flooding concerns. The minor administrative changes merely correct a table format inconsistency and update Basis section references.

Consequently, the operation of Surry Power Station with the proposed amendment and license condition will not:

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

The SW and CC Systems will function as designed under the Unit operating constraints specified by this project (i.e., Unit 2 in operation and Unit 1 in a refueling outage), and the potential for a loss of component cooling is already addressed by Station Abnormal Procedures. Therefore, there is no increase in the probability of an accident previously evaluated. The possibility of flooding due to failure of the temporary SW supply jumper in the Turbine Building basement has been evaluated and dispositioned by the implementation of appropriate precautions and compensatory measures to preclude damage to the temporary jumper and to respond to a postulated flooding event. A flood watch will be present around-the-clock with authority and procedural guidance to isolate the jumper, if required. Furthermore, the CCHXs serve no design basis accident mitigating function. Therefore, the consequences of an accident previously evaluated are not increased.

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

The SW and CC Systems' design functions and basic configurations are not being altered as a result of using a temporary SW supply jumper. The temporary jumper is designed to be safety-related and seismic with all of the design attributes of the normal SW supply line, except for the automatic isolation function and complete missile protection. The design functions of the SW and CC systems are unchanged as a result of the proposed changes due to (1) required plant conditions, (2) compensatory measures, (3) a Contingency Action Plan for restoration of the normal SW supply if required, and (4) strict administrative control of the temporary SW valve to preclude flooding or to isolate non-essential SW within the design basis assumed time limits. Unit 1 will be in a plant condition which will provide adequate time to restore the normal SW supply, if required. Therefore, since the SW and CC systems will basically function as designed and will be operated in their basic configuration, the possibility of a new or different type of accident than previously evaluated in the UFSAR [Updated Final Safety Analysis Report] is not created.

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

The margin of safety as defined in the Technical Specifications is not reduced since an operable SW flowpath to the required number of CCHXs is provided, and Unit operating constraints, compensatory measures and contingencies will be implemented as required to ensure the integrity and the capability of the SW flowpath. The use of the temporary jumper will be limited to the time period when missile producing weather is not expected, and Unit 1 meets specified unit conditions. Therefore, the temporary SW jumper, under the imposed project constraints and compensatory measures, provides the same reliability as the normal SW supply line. Furthermore, the Probabilistic Safety Assessment for Surry Power Station has been reviewed relative to potential flooding when the temporary SW jumper is in use. It has been determined that due to the SW restoration project's compensatory and contingency measures, as well as the constraints imposed by the Maintenance Rule online risk matrix, the impact on core damage frequency due to flooding is negligible.

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

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

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Project Director: P. T. Kuo, Acting.

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.

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

Date of amendment request: March 6, 1998.

Description of amendment request: The proposed amendment would (1) update the Technical Specification heatup and cooldown rate curves and extend their reactor fluence limit from the current 20 effective full power years (EFPY) to a new value of 35 EFPY, (2) incorporate into Technical Specifications the use of a Pressure and Temperature Limits Report (PTLR), and (3) change the power-operated relief valves (PORVs) temperature requirement for operability.

Date of individual notice in the Federal Register: March 27, 1998 (63 FR 14972).

Expiration date of individual notice: April 27, 1998.

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

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

Commonwealth Edison Company, Docket No. 50–237, Dresden Nuclear Power Station, Unit 2, Grundy County, Illinois

Date of application for amendment: March 19, 1998, as supplemented by letters dated March 28, 1998, and April 3, 1998.

Brief description of amendment: The amendment changes the Technical Specifications (TS) by revising the Dresden, Unit 2, Minimum Critical Power Ratio (MCPR) in TS Section 2.1.B and footnotes in TS Section 5.3.A, to allow the use of Siemens Power Corporation ATRIUM–9B fuel for all operating Modes at Dresden, Unit 2, Cycle 16.

Date of issuance: April 10, 1998.

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

Amendment No.: 168.

Facility Operating License No. DPR–19: The amendment revised the TS. Public comments requested as to proposed no significant hazards consideration: Yes (63 FR 14735 dated March 26, 1998). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by April 27, 1998, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment.

The March 28, 1998, and April 3, 1998, letters provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 10, 1998.

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

Local Public Document Room location: Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450.

Connecticut Yankee Atomic Power Company, Docket No. 50–213, Haddam Neck Plant Middlesex County, Connecticut

Date of application for amendment: Two applications, both dated May 30, 1997.

Brief description of amendment: Changes Administrative Controls Section of the Technical Specifications to implement new Certified Fuel Handler position and to implement revised management responsibilities and titles that reflect the permanently shut down status of the plant. In addition, minor typographical errors were corrected.

Date of issuance: March 27, 1998.

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

Amendment No.: 192.

Operating License No. DPR–61: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 16, 1997 (62 FR 38132 and 62 FR 38133). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 27, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Russell Library, 123 Broad Street, Middletown, CT 06457.

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

Date of application for amendment: March 26, 1997.

Brief description of amendment: The amendment revises the Containment Systems technical specifications (TS) to incorporate a note to allow opening an operable airlock door to perform repairs on inoperable airlock components when the other airlock door is inoperable. This amendment is in partial response to Consumers Energy's March 26, 1997, application. The Consumers Energy request also proposed revising the requirements contained in TS sections 3.6 and 4.5 to closely emulate the format and content of NUREG–1432, ''Standard Technical Specifications, Combustion Engineering Plants,'' (STS). That portion of the Consumers Energy request remains under staff review and will be addressed in a separate evaluation.

Date of issuance: April 8, 1998.

Effective date: April 8, 1998.

Amendment No.: 179.

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

Date of initial notice in **Federal Register:** December 17, 1997 (62 FR 66136).

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: January 28, 1998 (NRC–98–0008), as supplemented on March 10, 1998 (NRC–98–0036).

Brief description of amendment: The amendment revises the technical specifications (TSs) by modifying the ''#'' footnote to Table 1.2 and the ''*'' footnote to surveillance requirements 4.9.1.2 and 4.9.1.3 to permit the Reactor Mode Switch to be placed in the Run or Startup/Hot Standby positions to test switch interlock functions provided that all control rods are verified to remain fully inserted in core cells containing one or more fuel assemblies.

Date of issuance: March 31, 1998.

Effective date: March 31, 1998, with full implementation within 90 days.

Amendment No.: 116.

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

Date of initial notice in **Federal Register**: February 25, 1998 (63 FR 9599) The March 10, 1998, supplement requested a change in the implementation period and was not outside the scope of the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 31, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: December 17, 1997.

Brief description of amendments: The amendments revise Technical Specification Section 6.9.1.9 to reference updated or recently approved topical reports, which contain methodologies used to calculate cycle-specific limits contained in the Core Operating Limits Report. For several reports DEC indicated staff approval, but neglected to provide an ''A'' designation for the report number. Upon agreement by DEC, the staff has made these appropriate editorial corrections. These topical reports have all been previously approved by the staff under licensing actions separate from the current amendment request.

Date of issuance: April 8, 1998.

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

Amendment Nos.: Unit 1—178; Unit 2—160.

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

Date of initial notice in **Federal Register**: January 28, 1998 (63 FR 4311).

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

No significant hazards consideration comments received: No.

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

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

Date of amendment request: February 9, 1998.

Brief description of amendment: The amendment approves the use of the repair roll technology (reroll) for the upper tubesheet region of the ANO–1 steam generators. The reroll technology is an alternative to the either sleeving or plugging steam generator tubes found during inservice inspections to have defects that exceed the stated repair criteria. The reroll methodology works by creating a new mechanical tube to tubesheet structural joint below the tube defect indication.

Date of issuance: April 10, 1998.

Effective date: April 10, 1998.

Amendment No.: 190.

Facility Operating License No. DPR–51: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: February 24, 1998 (63 FR 9268).

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

No significant hazards consideration comments received: No.

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

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

Date of amendment request: April 1, 1998, as supplemented by letter dated April 8, 1998.

Brief description of amendment: The amendment allows approximately 440 steam generator tubes with confirmed volumetric indications within the upper tube sheet to remain in service during Cycle 15. The amendment revises TS 4.18.5.b to incorporate five criteria which need to be satisfied to allow steam generator tubes to remain in service during Cycle 15 with indications of outer diameter intergranular attack (ODIGA) in the upper tube sheet region of the steam generators.

Date of issuance: April 10, 1998.

Effective date: April 10, 1998.

Amendment No.: 191.

Facility Operating License No. DPR–51: The amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: No.

The Commission's related evaluation of the amendment, finding of emergency circumstances and final determination of no significant hazards consideration are contained in a Safety Evaluation dated April 10, 1998.

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

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

Date of amendment request: November 18, 1996, as supplemented by letter dated January 21, 1998.

Brief description of amendment: The amendment changes Technical Specification (TS) Surveillance Requirement 4.4.8.3.1.b to test the Shutdown Cooling System suction line relief valves in accordance with TS 4.0.5. Editorial changes to 4.4.8.3.1 and 4.4.8.3.1.a have also been made.

Date of issuance: April 1, 1998.

Effective date: April 1, 1998.

Amendment No.: 140.

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

Date of initial notice in **Federal Register**: February 11, 1998 (63 FR 6985).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 1, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of New Orleans

Library, Louisiana Collection, Lakefront, New Orleans, LA 70122.

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

Date of amendment request: May 24, 1997, as supplemented by letter dated January 21, 1998.

Brief description of amendment: The amendment modifies Technical Specifications (TS) 3.1.1.1, 3.1.1.2, 3.10.1 and Figure 3.1–1 by removing cycle dependent boron concentration and boration flow rate from the Action Statements and removing the ''RWSP at 1720 ppm'' curve from the figure. A change to TS Bases 3/4.1.1.1 and 3/4.1.1.2 has been included to support this change.

Date of issuance: April 8, 1998.

Effective date: April 8, 1998, to be implemented within 60 days.

Amendment No.: 141.

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

Date of initial notice in **Federal Register**: June 18, 1997 (62 FR 33123).

The January 21, 1998, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc.

Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: October 28, 1997, as supplemented by letter dated January 9, 1998.

Brief description of amendment: The amendment revised the Technical Specifications (TSs) to implement the containment leak rate testing provisions of 10 CFR Part 50, Appendix J, Option B.

Date of issuance: April 6, 1998.

Effective date: April 6, 1998.

Amendment No: 135.

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

Date of initial notice in **Federal Register**: December 3, 1997 (62 FR 63976).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 6, 1998.

No significant hazards consideration comments received: No.

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

IES Utilities Inc, Central Iowa Power Cooperative, and Corn Belt Power Cooperative, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: February 3, 1998.

Brief description of amendment: The amendment changes the operability requirement for the Standby Liquid Control system to Run/Power Operations and Startup.

Date of issuance: March 31, 1998.

Effective date: March 31, 1998.

Amendment No.: 221.

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

Date of initial notice in **Federal Register:** February 26, 1998 (63 FR 9874).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 31, 1998.

No significant hazards consideration comments received: No.

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

IES Utilities Inc, Central Iowa Power Cooperative, and Corn Belt Power Cooperative, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: October 3, 1997 as supplemented on December 10, 1997.

Brief description of amendment: The amendment revises the Operating License to allow the start of core offload as soon as 60 hours after shutdown.

Date of issuance: April 2, 1998.

Effective date: April 2, 1998.

Amendment No.: 222.

Facility Operating License No. DPR–49: Amendment revised the Operating License.

Date of initial notice in **Federal Register:** January 28, 1998 (63 FR 4314).

The December 10, 1997 submittal provided clarifying information that did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 2, 1998.

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: October 20, 1997, as supplemented February 10, and March 10, 1998.

Brief description of amendment: The amendment replaces in their entirety the existing Technical Specifications incorporated in Facility Operating License No. DPR–36 as Appendix A. Maine Yankee developed the revised Technical Specifications, titled Permanently Defueled Technical Specifications, to reflect the permanently shutdown and defueled status of the plant. Changes were made to the definitions, limiting conditions for operation, surveillance, and administrative control sections.

Date of issuance: March 30, 1998.

Effective date: March 30, 1998.

Amendment No.: 161.

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

Date of initial notice in **Federal Register**: December 3, 1997 (62 FR 63978). The February 10, and March 10, 1998, submittals added additional programs to the Section 5.5 Procedures and Section 5.6 Programs and Manuals did not change the proposed no significant hazards determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 30, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 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 31, 1997, as supplemented by letter dated February 3, 1998.

Brief description of amendment: This amendment changes Technical Specifications to support design changes to upgrade the analog-based average power range monitor system with General Electric's Nuclear Measurement Analysis and Control Power Range Neutron Monitor System, including an Oscillation Power Range Monitor function.

Date of issuance: March 31, 1998.

Effective date: As of the date of issuance to be implemented upon

completion and acceptance of design modifications resulting from the installation of the Nuclear Measurement Analysis and Control Power Range Neutron Monitor System.

Amendment No.: 80.

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

Date of initial notice in **Federal Register**: December 31, 1997 (62 FR 68310).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 31, 1998.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: September 26, 1997, as supplemented by letter dated March 12, 1998.

Description of amendment request: The amendment revises Technical Specification 3.7.6, ''Control Room Emergency Makeup Air and Filtration,'' and its associated Bases to separate the requirements for the control room air conditioning subsystem from the requirements for control room makeup air and filtration subsystem based on system function. The amendment also increases the allowed outage time for the Control Room Air Conditioning Subsystem.

Date of issuance: April 9, 1998.

Effective date: As of the date of issuance, with full implementation within 60 days.

Amendment No.: 56.

Facility Operating License No. NPF–86: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 22, 1997 (62 FR 54874). The March 12, 1998, supplemental letter did not change the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

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

Date of application for amendment: December 1, 1997.

Brief description of amendment: The amendment changes the Technical Specifications (TSs) by adding a 2.0 second plus or minus 0.1 second time delay to the 4.16 kV Emergency Bus Undervoltage Loss of Power, Level One, trip setpoint and allowable values in TS Table 3.3–4.

Date of issuance: April 1, 1998.

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

Amendment No.: 214.

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

Date of initial notice in **Federal Register:** January 14,1998 (63 FR 2280).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 1, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: November 11, 1997.

Brief description of amendment: The amendment allows NNECO to credit soluble boron for maintaining k-effective at less than or equal to 0.95 within the spent fuel pool rack matrix following a seismic event of a magnitude greater than or equal to an operating basis earthquake.

Date of issuance: April 9, 1998.

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

Amendment No.: 158.

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

Date of initial notice in **Federal Register:** December 3, 1997 (62 FR 63980).

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

No significant hazards consideration comments received: No.

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

Pacific Gas and Electric Company, Docket No. 50–133, Humboldt Bay Power Plant, Unit 3, Humboldt County, California

Date of application for amendment: December 9, 1996, as supplemented on June 12, 1997, and March 13, 1998.

Brief description of amendment: This amendment revised the Technical Specification to incorporate the requirements of appendix I of 10 CFR Part 50, into the Radiological Effluent Technical Specification (RETS) and to relocate the controls and limitations on RETS and radiological environmental monitoring (Currently in the Technical Specifications) to the Offsite Dose Calculation Manual and the Process Control Program. The amendment also revised the Technical Specifications to implement Generic Letter 89–10 (GL 89–10) and to incorporate the requirements of the revised 10 CFR Part 20.

Date of issuance: April 8, 1998.

Effective date: As of the date of issuance and shall be implemented no later than 30 days from the date of issuance.

Amendment No.: 32.

Facility Operating License No. DPR–7: Amendment revised the TS.

Date of initial notice in **Federal Register:** April 24, 1996 (61 FR 18174).

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

No significant hazards consideration comments received: No.

Local Public Document Room location: Humboldt County Library, 131 3rd Street, Eureka, California 95501.

Pennsylvania Power and Light Company, Docket No. 50–387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of application for amendment: August 26, 1997, as supplemented by letters dated December 4, 1997, February 2, March 19, and April 2, 1998.

Brief description of amendment: This amendment changes the Susquehanna Unit 1 Technical Specifications to support the use of the Siemens Power Corporation ATRIUM–10 fuel design in the upcoming Cycle 11 refueling outage.

Date of issuance: April 6, 1998.

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

Amendment No.: 174.

Facility Operating License No. NPF–14: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 31, 1997 (62 FR 68314).

The December 4, 1997, February 2, March 19, and April 2, 1998, submittals provided clarifying information that did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 6, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: September 2, 1997.

Brief description of amendment: These amendments revise LGS, Units 1 and 2, TS Section 4.0.5, and Bases Sections B 4.0.5 and B 3/4.4.8 regarding the surveillance requirements associated with Inservice Testing and Inservice Inspection Programs of the American Society of Mechanical Engineers Code Class 1, 2, and 3 components.

Date of issuance: March 31, 1998.

Effective date: March 31, 1998.

Amendment Nos.: 125 and 89.

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

Date of initial notice in **Federal Register:** February 11, 1998 (63 FR 6990).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 31, 1998.

No significant hazards consideration comments received: No.

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

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

Date of amendments request: July 23, 1997, as supplemented by letters dated September 30, October 27, and December 18, 1997, and February 12, 1998.

Brief Description of amendments: The amendments revise the Pressure-Temperature Limit Heatup, Cooldown, and Hydrostatic Testing curves for Farley Units 1 and 2 and relocate the curves from the Technical Specifications to a Pressure and Temperature Limits Report for each unit.

Date of issuance: April 9, 1998.

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

Amendment Nos.: Unit 1–136; Unit 2–128.

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

Date of initial notice in **Federal Register**: September 10, 1997 (62 FR 47699); January 14, 1998 (63 FR 2281); February 23, 1998 (63 FR 9020).

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

No significant hazards consideration comments received: No.

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

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

Date of amendment request: October 24, 1997 (TXX–97228).

Brief description of amendments: The amendments revise core safety limit curves and Overtemperature N–16 reactor trip setpoints based on analyses of the core configuration and expected operation for CPSES Unit 1, Cycle 7. The changes apply equally to CPSES Units 1 and 2 licenses since the Technical Specifications are combined.

Date of issuance: March 27, 1998.

Effective date: March 27, 1998, to be implemented within 30 days.

Amendment Nos.: Unit 1—Amendment No. 57; Unit 2—Amendment No. 43.

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

Date of initial notice in **Federal Register**: November 19, 1997 (62 FR 61847).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 27, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 8, 1997, as supplemented by letter dated January 16, 1998.

Brief description of amendment: The amendment revises the Callaway Plant, Unit 1 Technical Specification Table 3.3–3 Functional Units 4.b.2 and 5.a.2 to make the number of main steam and feedwater isolation system (MSFIS) channels consistent with the solid state protection system, adds a clarifying note and changes Table 4.3–2 Functional Units 4.b.2 and 5.a.2 slave relay quarterly test to a monthly staggered actuation logic test.

Date of issuance: March 25, 1998.

Effective date: March 25, 1998, to be implemented within 30 days from the date of issuance.

Amendment No.: 123.

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

Date of initial notice in **Federal Register**: December 17, 1997 (62 FR 66143) The January 16, 1998, supplemental letter provided additional clarifying information that did not change the staff's original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 25, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 17, 1997, as supplemented by letters dated March 3, 1998, and March 17, 1998.

Brief description of amendment: The amendment revises the technical specifications to modify the heatup and cooldown curves and the maximum allowable power operated relief valve setpoint curves for cold overpressure protection.

Date of issuance: April 2, 1998.

Effective date: April 2, 1998, to be implemented within 30 days from the date of issuance.

Amendment No.: 124.

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

Date of initial notice in **Federal Register**: January 14, 1998 (63 FR 2282).

The March 3, 1998, and March 17, 1998, supplemental letters provided

additional 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 April 2, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 11, 1996.

Brief description of amendment: The amendment revises the Technical Specifications regarding the amount of foam concentrate required to support operability of the reactor recirculation motor generator set foam fire suppression system.

Date of Issuance: March 31, 1998.

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

Amendment No.: 156.

Facility Operating License No. DPR–28: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 22, 1997 (62 FR 54877).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated March 31, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

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

Date of application for amendment: June 9, 1997.

Brief description of amendment: The amendment revises Technical Specification Section 6.0 to add and revise reference to NRC-approved methodologies which will be used to validate or generate the cycle-specific thermal hydraulic stability based operating limits in the Vermont Yankee Core Operating Limits Report.

Date of Issuance: April 7, 1998.

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

Amendment No.: 157.

Facility Operating License No. DPR–28: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 13, 1997 (62 FR 43377).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated April 7, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

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

Date of application for amendment: August 22, 1996.

Brief description of amendment: The amendment revises the Technical Specifications to change the action statement for the high range stack noble gas monitor based on the guidance of Generic Letter 83–36, NUREG–0737 Technical Specifications.

Date of Issuance: April 8, 1998.

Effective date: April 8, 1998, to be implemented within 30 days.

Amendment No.: 158.

Facility Operating License No. DPR–28: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 4, 1997 (62 FR 30647).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated April 8, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

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

Date of amendment request: January 28, 1998.

Brief description of amendment: The amendment revises Technical Specification §§ 6.3 and 6.12 to reflect a merger for the positions of Superintendent Radiation Protection and Superintendent Chemistry into one new position, Manager Chemistry/Radiation Protection.

Date of issuance: March 30, 1998.

Effective date: March 30, 1998.

Amendment No.: 115.

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

Date of initial notice in **Federal Register:** February 25, 1998 (63 FR 9614).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 30, 1998.

No significant hazards consideration comments received: No.

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

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

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 for the amendment 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.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an

opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective 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 application for amendment, (2) the amendment to Facility Operating License, 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 room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By May 22, 1998, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee. Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

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

Date of amendment request: April 2, 1998 (NRC–98–0062).

Description of amendment request: The amendment revised the action specified in Technical Specification Table 3.3.7.5–1 if one channel of drywell oxygen monitoring is inoperable.

Date of issuance: April 3, 1998.

Effective date: April 3, 1998, with full implementation by April 6, 1998.

Amendment No.: 117.

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

Public comments requested as to proposed no significant hazards consideration: No. The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated April 3, 1998.

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

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

NRC Project Director: Cynthia A. Carpenter.

Dated at Rockville, Maryland, this 15th day of April 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 98–10470 Filed 4–21–98; 8:45 am]

BILLING CODE 7590–01–P

RAILROAD RETIREMENT BOARD

Proposed Collection; Comment Request

SUMMARY: In accordance with the requirement of Section 3506 (c)(2)(A) of the Paperwork Reduction Act of 1995 which provides opportunity for public comment on new or revised data collections, the Railroad Retirement Board (RRB) will publish periodic summaries of proposed data collections.

Comments are invited on: (a) Whether the proposed information collection is necessary for the proper performance of the functions of the agency, including whether the information has practical utility; (b) the accuracy of the RRB's estimate of the burden of the collection of the information; (c) ways to enhance the quality, utility, and clarity of the information to be collected; and (d) ways to minimize the burden related to the collection of information on respondents, including the use of automated collection techniques or other forms of information technology.

Title and purpose of information collection: Employer Service and Compensation Reports; OMB 3220–0070 Section 2(c) of the Railroad Unemployment Insurance Act (RUIA) specifies the maximum normal unemployment and sickness benefits that may be paid in a benefit year. Section 2(c) further provides for extended benefits for certain employees and for beginning a benefit year early for other employees. The conditions for these actions are prescribed in 20 CFR 302.

All information about creditable railroad service and compensation needed by the RRB to administer Section 2(c) is not always available from annual reports filed by railroad employers with the RRB (OMB 3220–0008). When this occurs, the RRB must obtain supplemental information about service and compensation.

The RRB utilizes Form UI–41, Supplemental Report of Service and Compensation, and Form UI–41a, Supplemental Report of Compensation, to obtain the additional information about service and compensation from railroad employers. Completion of the forms is mandatory. One response is required of each respondent. The RRB proposes to revise Form UI–41 and UI–41a to add language required by the Paperwork Reduction Act of 1995. Minor editorial changes are also proposed. The completion time for Form UI–41 and UI–41a is estimated at 8 minutes per response.

ADDITIONAL INFORMATION OR COMMENTS: To request more information or to obtain a copy of the information collection justification, forms, and/or supporting material, please call the RRB Clearance Officer at (312) 751–3363. Comments regarding the information collection should be addressed to Ronald J. Hodapp, Railroad Retirement Board, 844 N. Rush Street, Chicago, Illinois 60611–2092. Written comments should be received on or before June 22, 1998.

Chuck Mierzwa,

Clearance Officer.

[FR Doc. 98–10586 Filed 4–21–98; 8:45 am]

BILLING CODE 7905–01–M

SECURITIES AND EXCHANGE COMMISSION

[Release No. 35–26857]

Filings Under the Public Utility Holding Company Act of 1935, as amended (''Act'')

April 16, 1998.

Notice is hereby given that the following filing(s) has/have been made with the Commission pursuant to provisions of the Act and rules promulgated under the Act. All interested persons are referred to the application(s) and/or declaration(s) for complete statements of the proposed transaction(s) summarized below. The application(s) and/or declaration(s) and any amendments is/are available for public inspection through the Commission's Office of Public Reference.

Interested persons wishing to comment or request a hearing on the application(s) and/or declaration(s) should submit their views in writing by May 12, 1998, to the Secretary, Securities and Exchange Commission, Washington, D.C. 20549, and serve a copy on the relevant applicant(s) and/or declarant(s) at the address(es) specified below. Proof of service (by affidavit or, in case of an attorney at law, by certificate) should be filed with the request. Any request for hearing should identify specifically the issues of fact or law that are disputed. A person who so requests will be notified of any hearing, if ordered, and will receive a copy of any notice or order issued in the matter. After May 12, 1998, the application(s) and/or declaration(s), as filed or as amended, may be granted and/or permitted to become effective.

American Electric Power Company, Inc. and *Central and South West Corporation* (70–9169).

Notice of Proposal to Amend Certificate of Incorporation to Increase Number of Authorized Shares of Common Stock; Order Authorizing Solicitation of Proxies.

American Electric Power Company, Inc. (''AEP''), 1 Riverside Plaza, Columbus, Ohio 43215, and Central and South West Corporation (''CSW''), 1616 Woodall Rodgers Freeway, Dallas, Texas 75266, each a registered holding company, have filed a joint declaration with this Commission under sections 6(a)(2), 7 and 12(e) of the Public Utility Holding Company Act of 1935, as amended (''Act''), and rules 62 and 65 under the Act.

AEP and CSW have entered into an Agreement and Plan of Merger, dated as of December 21, 1997 (''Merger Agreement''). Under the Merger