December 22, 1993

IDENTICAL LETTERS SENT TO: (See attached list of addressees)

The Honorable Joseph Lieberman, Chairman
Subcommittee on Clean Air and Nuclear Regulation
Committee on Environment
and Public Works
United States Senate
Washington, DC 20510

Dear Mr. Chairman:

Public Law 97-415, enacted on January 4, 1983, amended Section 189 of the Atomic Energy Act of 1954 to authorize the Nuclear Regulatory Commission 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.

In addition, the legislation requires the Commission to periodically (but not less frequently than once every 30 days) publish notice of any amendments issued, or proposed to be issued, under the new authority above.

Enclosed for your information is a copy of the Commission's Biweekly Notice of Applications and Amendments to Operating Licenses involving no significant hazards considerations, which was published in the Federal Register on December 8, 1993 (58 FR 64598).

Sincerely,

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosure:
Federal Register
Notice

cc: Senator Alan K. Simpson

002083

DISTRIBUTION w/o enclosure:

Central File	NRC PDR
OGC	OCA (3)
JTaylor	TMurley/FMiraglia
SECY Mail Facility	

RETURN CONCURRENCE PAGE TO:
Doris Twigg 14-H-25

LA:PD23	D:DRP	ADP	DD:NRR	D:NRR	OCA
LBerry	SVarga	LJCallan	FMiraglia	TMurley	
12/16/93	12/ /93	12/17/93	12/ /93	12/20/93	12/21/93

130023

The Honorable Richard H. Lehman, Chairman
Subcommittee on Energy and Mineral Resources
Committee on Natural Resources
United States House of Representatives
Washington, DC 20515

cc: Representative Barbara Vucanovich

The Honorable Philip R. Sharp, Chairman
Subcommittee on Energy and Power
Committee on Energy and Commerce
United States House of Representatives
Washington, DC 20515

cc: Representative Michael Bilirakis

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 November 15, 1993, through November 26, 1993. The last biweekly notice was published on November 24, 1993 (58 FR 62149).

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

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

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

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

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room P-223, Phillips Building, 7920 Norfolk Avenue, Bethesda, 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 20555. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By January 7, 1994, 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 20555 and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555, and at the local public document room for the particular facility involved.

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

Date of amendments request: November 2, 1993

Description of amendments request: The proposed amendments would revise the Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Technical Specifications (TSs) regarding surveillance requirements associated with the emergency diesel generators (EDGs). The EDGs are used to provide electrical power for the operation of Engineered Safety Features (ESF) and safe shutdown equipment for events involving a loss of offsite power. Should a loss of power be sensed on one of the 4160 volt ESF busses, the EDGs will automatically start and power equipment needed to safely shut the Unit down. If an accident condition is present, the EDG will start, but will only supply power to the ESF busses if offsite power is lost.

Specifically, the requested changes are:

1. TS 4.8.1.1.2.d - This change to the TSs extends the interval from 18 months to the current refueling interval of 24 months for the surveillances listed under 4.8.1.1.2.d. The provisions of Specification 4.0.2 would continue to apply to this specification.

2. TS 4.8.1.1.2.a.4 - This change removes the requirement to verify a

specific EDG speed of 900 revolutions per minute (rpm). The requirement to verify the frequency assures that the proper speed is achieved.

3. TS 4.8.1.2 - This change adds the EDG surveillances dealing with sequencer testing to the list of surveillances that can be exempted in Modes 5 and 6.

4. TS 4.8.1.1.2.d.5 - This change eliminates the specific numerical reference of 2700 kW associated with the 2000 hour rating of an EDG being tested.

5. TS 4.8.1.1.2.c and 4.8.1.1.2.d.3.b - This change will allow the EDGs to be pre-lubricated prior to being started which is in accordance with vendor recommendations.

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

The Calvert Cliffs Emergency Diesel Generators (EDGs) are used to provide electrical power for the operation of Engineered Safety Features (ESF) and safe shutdown equipment for events involving a loss of offsite power. Should a loss of power be sensed on one of the 4160 volt ESF busses, the EDGs will automatically start and power equipment needed to safely shut the Unit down. If an accident condition is present, the EDG will start, but will only supply power to the ESF busses if offsite power is lost.

The proposed changes will modify several Technical Specification Surveillances associated with testing of the EDGs.

Technical Specification 4.8.1.1.2.d verifies the overall condition of the EDG is acceptable. A major maintenance inspection and several tests involving starting and loading the EDG are performed every 18 months (old refueling interval) in accordance with the surveillance. An evaluation was conducted to determine if the surveillance interval could be extended from 18 months to 24 months (current refueling interval). The evaluation concluded there were no problems attributed to time dependence. Extending the interval to 24 months will eliminate the need for a special outage after 18 months, thus eliminating the possibility of encountering plant transients associated with a plant shutdown and startup. Extending the surveillance interval to 24 months will not significantly increase the probability of the EDG failing to operate as assumed in previously evaluated accidents.

Additionally the EDGs are not initiators to any previously evaluated accident. Therefore, extending the surveillance interval will not increase the consequences of an accident previously identified.

Two of the requested surveillance changes remove specific values and do not alter the intent of the surveillances. Technical Specification 4.8.1.1.2.a.4 verifies the EDG reaches 900 rpm rated speed after being started. Speed and frequency are directly related and the critical parameters that should be monitored closely are frequency and voltage. Removal of the specific value for speed will have no effect on surveillance results. Technical Specification 4.8.1.1.2.d.5 verifies the auto-connected accident loads powered by the EDG do not exceed the EDGs' 2000 hour capacity rating. Modifications to increase the EDGs' capacity will be performed in future outages. To reflect this capacity change, the current value of 2700 kW listed in the Technical Specification should be removed. The actual surveillance steps and intent will remain unchanged. Therefore, these changes would have no effect on the probability or consequences of an accident previously evaluated.

The Technical Specifications require two EDGs to be operable in Modes 1-4, and one EDG in Modes 5 and 6. The EDG surveillances performed in Modes 5 and 6 are identical to those performed in Modes 1-4, yet plant conditions are quite different. The instrumentation that detects a loss of voltage on the 4160 volt busses is not required in Modes 5 and 6 and much of the ESF equipment is not required to be operable. The proposed change would modify Technical Specification 4.8.1.2 to reflect the status of plant conditions and equipment when the unit is shutdown. The EDG loss-of-coolant incident sequencer which is designed to load ESF and equipment needed to safely shutdown the plant do not need to be tested when the unit is already shutdown. The undervoltage instrumentation signals required to initiate sequencer action are not credited in the Updated Final Safety Analysis Report (UFSAR) for events which occur during shutdown modes. Therefore, eliminating sequencer testing for operability in the shutdown modes will have no effect on the probability or consequences of accidents previously evaluated.

Emergency Diesel Generator reliability and availability will be maintained if wear and stress are reduced when the EDGs are started. Proper warm-up and pre-lubrication techniques, as recommended by the vendor, will help minimize the potential for degradation. Reliable EDG starts due to actual losses of power on 4160 volt busses prove their capability to perform their required safety function.

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

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

The proposed changes do not represent a significant change in the configuration or operation of the plant.

These changes represent clarifications and improvements to the Technical Specification surveillances only and do not affect assumptions associated with the EDGs in the UFSAR. The changes will modify surveillance requirements such as the verification of a specific value (900 rpm, 2700 kW) and frequency of the surveillance (18 to 24 months, Modes 5 and 6 testing). The changes will not alter the intent or method in which the surveillance is conducted.

Allowing pre-lubrication for planned fast starts does change the current test method, but will help maintain EDG reliability.

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

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

The proposed changes do not affect the margin of safety credited to the EDG function. The EDG will continue to provide power to ESF and safe shutdown components as stated in the UFSAR. The availability of the EDGs will not be reduced by these changes and the intent of the surveillances will be preserved.

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 amendments request involves no significant hazards consideration.

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

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

NRC Project Director: Robert A. Capra

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

Date of amendments request: November 3, 1993

Description of amendments request: The proposed amendments would revise the Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Technical Specifications (TSs) by modifying the surveillance requirements to reflect the removal of the auto-closure interlock (ACI) function from the shutdown cooling (SDC) system. The SDC system is used to achieve and maintain the reactor coolant system (RCS) in cold shutdown by removing decay heat from the reactor core following shutdown of the reactor. The ACI is designed to provide a close signal to the SDC system suction isolation valves when the RCS pressure exceeds a predetermined pressure setpoint. A generic evaluation demonstrated that removing the ACI function and replacing it with a valve position alarm will reduce the number of spurious closures of the SDC system suction isolation valves which in turn will increase the system availability and result in an overall decrease in shutdown risk. The generic evaluation was supplemented by a plant specific evaluation for the Calvert Cliffs facility which provided the same results.

The proposed amendments also revise the setpoint for the open permissive interlock (OPI) which is designed to prevent opening of the SDC system suction isolation valves when the RCS pressure is above the setpoint. The proposed setpoint is based on the pressurizer pressure at the instrument tap and accounts for instrument uncertainties.

Specifically, TS 3/4.5.e.1 is changed to require verification that the OPI prevents the SDC system suction valves from being opened when the RCS pressure is greater than or equal to 309 psia. The requirement to verify the automatic isolation (the ACI function) is deleted. The TS Bases Section B 3/4 5.2 is changed to reflect the proposed changes discussed above.

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

The Calvert Cliffs Shutdown Cooling (SDC) System is used to achieve and maintain the Reactor Coolant System (RCS) in cold shutdown condition by removing the decay heat from the reactor core following plant shutdown. The Auto-Closure Interlock (ACI) is designed to provide a close signal to the SDC System suction isolation valves when the RCS pressure exceeds the predetermined pressure setpoint. This proposed change would modify Technical Specification Surveillance Requirement 4.5.2.e.1 to reflect removal of the ACI function. The Open Permissive Interlock (OPI), which is designed to prevent opening of the SDC System suction isolation valves when RCS pressure is above the pressure setpoint, would remain. The removal of ACI was evaluated generically in the report CE NSPD-550 in terms of the availability of the SDC System. This generic evaluation has been supplemented by a plant-specific evaluation for Calvert Cliffs. The evaluation demonstrated that removing ACI and replacing it with a valve position alarm will reduce the number of spurious closures of the SDC System suction isolation valves and thus increase the availability of the SDC System, resulting in a corresponding decrease in shutdown risk. Revising the OPI action from 300 psia to 309 psia is a result of establishing a clear basis for this value. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The report CE NSPD-550 also evaluated the removal of the SDC System ACI in terms of the frequency of an inter-system Loss of Coolant Accident (ISLOCA) and the effect on overpressure transients. The plant-specific evaluation for Calvert Cliffs showed a negligible change in the calculated probability of an ISLOCA event associated with ACI removal. The proposed change to remove the ACI surveillance requirement and the setpoint change will not alter the effect of an overpressure transient at cold shutdown conditions. The ACI was intended to ensure that the SDC System is properly isolated from RCS pressure during start-up operations. The ACI function does not protect against a malfunction of the valve which results in its failure to close. The valve position alarm will warn the operator of a failure to manually close the valve as well as a valve malfunction. While it is true that the ACI initiates an auto-closure of the SDC System suction isolation valves on high RCS pressure, overpressure protection of the SDC System is provided by the SDC System relief valve and administrative controls, and not by the slow-acting suction isolation valves that isolate the SDC System from the RCS. The possibility of a loss of SDC System is reduced by the proposed change because the potential of the SDC System suction isolation valves being closed by a spurious signal will be eliminated. No other failures are introduced by removing ACI. [Also, revising the OPI action does not introduce a new or different type of accident.] Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

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

The ACI function is not credited in a margin of safety for any accident previously evaluated and is not discussed in the basis for Technical Specification 3/4.5.2. The ACI function is intended to provide a backup to the operator action of closing the SDC System suction isolation valves during plant pressurization. The evaluation of CE NSPD-550 and the Calvert Cliffs plant-specific evaluation indicates that the availability of the SDC System is increased with removal of ACI. In place of ACI, the installation of new visual and audible alarms in the control room, along with procedural changes and operator training, will reduce shutdown risk for the plant by eliminating the possibility of a spurious signal closing the SDC System suction isolation valves during shutdown cooling operation. Revising the OPI action limit is a result of establishing a clear basis for this value. 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 amendments request involves no significant hazards consideration.

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

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

NRC Project Director: Robert A. Capra

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

Date of amendments request: November 3, 1993

Description of amendments request: The proposed amendments would revise the Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Technical Specifications (TSs) by eliminating the TSs that are applicable to the incore instrument (ICI) system. The limitations on the use of the ICI system will be relocated to the Updated Final Safety Analysis Report (UFSAR). The ICI system is used to measure core power distribution for the purpose of monitoring the TS limits on Linear Heat Rate, Total Planar Radial Peaking Factor, Total Integrated Radial Peaking Factor, and Azimuthal Power Tilt. The ICI system has no safety purpose itself; it only measures values which have safety significance. No change to the monitored values is proposed. The proposed change will relocate requirements on the number and distribution of incore detectors used by the ICI system when measuring these values from the TSs to the UFSAR. The licensee has determined that the requirements on the ICI system are not constraints on design and operation which belong in the TSs. In addition, NUREG-1432, "Standard Technical Specifications for Combustion Engineering Plants," does not include TS requirements on the ICI system.

Specifically, the following changes are proposed:

(1) TS 3/4.3.3, which provides the requirements for the incore detectors is deleted.

(2) TS 3/4.2.1.4.b is revised to remove uncertainty factors which are applied to the ICI system.

(3) TSs 3.2.1, 4.2.1.4, 3.2.2.1, 4.2.2.1, 3.2.3 and 4.2.3.2.b are revised to remove the cycle specific foot notes.

(4) The Table of Contents and TS Bases Section 3/4.3 are revised to reflect the proposed 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. Would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Incore Instrument (ICI) System is used to measure core power distribution for the purpose of monitoring the technical specification limits on Linear Heat Rate, Total Planar Radial Peaking Factor, Total Integrated Radial Peaking Factor, and Azimuthal Power Tilt. The ICI System has no safety purpose itself; it only measures values which have safety significance. No change to the monitored values is proposed. The proposed change will relocate requirements on the number and distribution of incore detectors used by the ICI System when measuring these values from the Technical Specifications to the Updated Final Safety Analysis Report (UFSAR). This will allow changes to the requirements to be made without Commission approval as long as the changes meet the criteria of 10 CFR 50.59. Changes to the ICI System requirements which do not meet the criteria of 10 CFR 50.59 must be approved by the Commission by license amendment.

Relocation of the requirements on the ICI System from the Technical Specifications to the UFSAR does not increase the probability or consequences of any accident previously analyzed because the ICI System is neither a precursor or a mitigator for any analyzed accident. The ICI System is not credited in any safety analysis. The values measured by the ICI System are important parameters in many accident analyses, however, this proposed change does not remove or affect the limits on these values.

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

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

The proposed change does not represent a change in the configuration or operation of the plant. The ICI System will continue to be used to monitor Technical Specification limits on core power distribution. The core power distribution Technical Specification limits are not changed.

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

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

The ICI System makes no contribution to the margin of safety. The ICI System is used to measure core power distribution values which do contain a margin of safety. The limits on these values are not changed.

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 amendments request involves no significant hazards consideration.

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

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

NRC Project Director: Robert A. Capra

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

Date of amendments request: November 5, 1993

Description of amendments request: The proposed amendment consists of two related changes. The first change modifies the Calvert Cliffs containment penetration technical specifications (TSs) to resemble the containment penetration TS in NUREG-1432, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors" (STS). The second change allows the containment personnel airlock to be open during fuel movement and core alterations.

Specifically, the first change revises Specification 3.9.4, "Containment Penetrations, Shutdown" to make it consistent with the same Specification in the STS. It deletes "positive reactivity changes" and "movement of heavy loads over irradiated fuel within the containment building" from the Applicability, Actions, and Surveillance sections. In addition, the Applicability and Surveillance sections are revised by removing references to "degraded electrical conditions" and substituting equivalent actions in lieu of references to Specification 3.9.4 in Specifications 3.8.1.2, 3.8.2.2, and 3.8.2.4.

The second change revises Specification 3.9.4, "Containment Penetrations, Shutdown," to allow the containment personnel airlock (PAL) to be open during fuel movement and core alterations provided that one PAL door is operable, the plant is in MODE 6 with 23 feet of water bove the fuel, and a designated individual is continuously available to close the airlock door. This individual must be stationed at the Auxiliary Building side of the outer airlock door.

Consistent with STS, features required for PAL operability are given in the Bases. The Bases state that in order for a PAL door to be operable, it must be capable of being closed and the airlock doorway must not be blocked. In addition, Specification 3.9.3, "Decay Time," is modified to lengthen the minimum time between subcriticality and fuel movement from 72 hours to 100 hours.

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:

Change 1 - Modify The Calvert Cliffs Containment Penetration Technical Specifications To Resemble The Standard Technical Specifications

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

The only previously evaluated accident affected by containment penetration status during shutdown is a fuel handling accident in the containment. Containment penetration closure is required during periods when the plant is shutdown and the risk of a fuel handling accident is higher in order to minimize the release of radioactive material due to such an accident. The proposed change modifies the conditions of Specification 3.9.4 regarding when containment penetration closure is required in order to make the Calvert Cliffs technical specification resemble the Standard Technical Specifications, Combustion Engineering Plants (NUREG-1432). This involves eliminating applicability of the specification during periods of positive reactivity addition, movement of heavy loads over irradiated fuel in the containment, and periods of electrical degradation. Containment penetrations are not an initiator to any accident so the status of containment penetrations has no affect on the probability of an accident previously evaluated.

Two applicability conditions of Specification 3.9.4, "positive reactivity additions" and "movement of heavy loads of irradiated fuel in the containment," are not needed because equivalent protection is provided by Specifications 3.9.1, "Refueling Boron Concentration," and by previous analysis of control of heavy loads. The actions to be taken during electrical degradation have been relocated from Specification 3.9.4 to the electrical specifications (Technical Specifications 3.8.1.2, 3.8.2.2, and 3.8.2.4). Therefore, the proposed changes provide a level of protection against radioactive release from the containment during shutdown conditions equivalent to the existing specifications.

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

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

The proposed changes to Specification 3.9.4 will provide a level of protection against radioactive release from the containment equivalent to the current specifications. It does not represent a significant change in the configuration or operation of the plant which could create the possibility of a new type of accident. Positive reactivity changes which could potentially violate the required shutdown margin were evaluated in determining the technical specification limit for refueling boron concentration (Specification 3.9.1), and movement of heavy loads over irradiated fuel has been previously evaluated in our response to NUREG-0612, "Control of Heavy Loads." Actions to be taken during periods of electrical degradation have been relocated within the technical specifications but remain unchanged.

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

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

The proposed changes will eliminate some conditions when containment penetration closure is required. This could allow the release of radioactivity from containment. However, for each eliminated condition there is an existing equivalent or more restrictive requirement which would prevent events which would result in a radioactive release. Therefore, there will be no increase in offsite dose and the margin of safety is maintained.

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

Change 2 - Modify The Calvert Cliffs Containment Penetration Technical Specifications To Allow The Containment Personnel Airlock To Be Open During Fuel Movement And Core Alterations

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

The proposed change to Specification 3.9.4 would allow the containment personnel airlock (PAL) to be open during fuel movement and core alterations. The PAL is closed during fuel movement and core alterations to prevent the escape of radioactive material in the event of a fuel handling accident. The PAL is not an initiator to any accident. Whether the PAL doors are open or closed during fuel movement and core alterations has no affect on the probability of any accident previously evaluated.

Allowing the PAL doors to be open during fuel movement and core alterations does increase the consequences of a fuel handling incident in the containment from no offsite dose to 14.06 Rem to the thyroid and 0.457 Rem to the whole body. However, the calculated offsite doses are less than 5% of the limits of 10 CFR Part 100 and, therefore, do not represent a significant increase in offsite dose. In addition, the calculated doses are larger than the expected doses because the calculation does not incorporate the closing of the PAL door after the containment is evacuated. The proposed change will significantly reduce the dose to workers in the containment in the event of a fuel handling accident by speeding the containment evacuation process. The proposed change will also significantly decrease the wear on the PAL doors and, consequently, increase the availability of the PAL doors in the event of an accident.

The proposed change increases the minimum decay time from shutdown to the movement of irradiated fuel in containment. Minimum decay time is not a precursor to any accident. Lengthening the minimum decay time decreases the consequences of a fuel handling accident by reducing the radioactive inventory of the irradiated fuel.

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

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

The proposed change affects a previously evaluated accident, e.g., a fuel handling incident. It does not represent a significant change in the configuration or operation of the plant and, therefore, does not create the possibility of a new or different type of accident from any accident previously evaluated.

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

The margin of safety as defined by 10 CFR Part 100 has not been significantly reduced. There is an increase in calculated offsite dose resulting from a fuel handling accident but the increase is a small fraction of the limits given in 10 CFR Part 100. The proposed change also increases the minimum decay time from shutdown to the movement of irradiated fuel in containment. This change reduces the offsite dose in the event of a fuel handling accident which partially compensates for the higher offsite doses under this proposed change. 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 amendments request involves no significant hazards consideration.

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

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

NRC Project Director: Robert A. Capra

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

Date of amendment request: October 19, 1993

Description of amendment request: The proposed amendment would remove the Low Condenser Vacuum Scram (LCVS) and reduce the turbine first stage pressure setpoint at which it is permissible to bypass the turbine control valve fast closure and the turbine stop valve closure trip (scram) signals.

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

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

REMOVAL OF LOW VACUUM SCRAM

The LCVS is not required to ensure the safe operation of Pilgrim Station. The LCVS is provided to anticipate the reactor scram associated with the turbine trip caused by low condenser vacuum. (Reference: "PNPS Final Safety Analysis Report," Section 7.2.3.8) and is not relied upon in the plant transient analysis. PNPS [Pilgrim Nuclear Power Station] FSAR [Final Safety Analysis Report], Section R.2.1.2 explains that an instantaneous loss of vacuum is the most severe vacuum transient and is equivalent to a turbine trip without bypass. Slow vacuum transients allow for some bypass steam flow (the bypass shuts at 7 inches of vacuum) and thus results in less severe transients. In addition, the "PNPS Reload Analysis" (NEDE-24011-P-A-4-US, Standard Application for Reactor Fuel) does not take credit for LCVS. PNPS FSAR, "Section 14.4" includes low vacuum transients under turbine trip without bypass. Since this turbine trip scram will remain, and since the LCVS is intended to anticipate the turbine trip scram as well as not being a distinct element of the accident analysis, instrumentation associated with the LCVS will be removed from Pilgrim and the scram will no longer exist. Removal of the LCVS from Technical Specifications and from Pilgrim will not result in a significant increase in the probability or consequences of an accident previously evaluated but will reduce the possibility of spurious scrams.

REVISION OF TURBINE FIRST STATE PRESSURE SCRAM SETPOINT

The scram signal generated by closure of the TSVs [turbine stop valves] or fast closure of the TCVs [temperature control valves] preserve sufficient thermal margin for pressurization transients at high core thermal powers. At core thermal powers below 45% of rated, the severity of pressurization transients is reduced such that these scram signals are no longer required, the reactor high-pressure and high-flux scram setpoints provide protection for the reactor as described in PNPS FSAR Section 7.2. These scram signals are bypassed in the interest of improved plant availability when thermal margin considerations permit.

The Pilgrim Reactor Protection (RPS) uses the high-pressure turbine section first-stage bowl pressure rather than core thermal power to determine when the scram signals generated by closure of the TSVs or fast closure of the TCVs can be bypassed. Turbine bowl pressure is proportional to core thermal power and is also related to the balance-of-plant (BOP) configuration. Therefore, the maximum bowl pressure above which the scram signals cannot be bypassed must correspond to 45% of rated core thermal power for the most limiting balance-of-plant configuration.

A reduction in the degree of feedwater heating results in a decreased turbine bowl pressure for a particular core thermal power. Hence, the limiting balance-of-plant configuration for this evaluation assumes all feedwater heaters are out-of-service. In addition to the degree of feedwater heating, the bowl pressure is also affected by the amount of turbine bypass flow. Bypassing flow around the turbine further reduces the bowl pressure for a particular core thermal power and feedwater heater configuration. However, General Electric analysis, "EAS-53-0587, Rev. 1", shows the limiting balance-of-plant configuration does not need to consider opened turbine bypass valves, because, with these valves opened, the consequences of design-basis transients are acceptable without scram signals being generated upon closure of the TSV's or fast closure of the TCV's, even at core thermal powers greater than 45% of rated.

Based on the above considerations and to provide added conservatism to minimize the possibility of lifting the SRV's after a Turbine Trip at low power, the maximum turbine first stage bowl pressure permitting scram signal bypass is determined to be 112 psig. Changing the currently allowed maximum of 305 psig to 112 psig brings the specified datum into conformance with Pilgrim's design and, thereby, does not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Pilgrim Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any acciden previously evaluated.

LCVS is not part of the Pilgrim Station design basis. Its removal from technical specifications does not, therefore, present any new or different challenges to the integrity or responses of systems designed to prevent or mitigate an accident. Hence, the removal of LCVS from technical specifications and from Pilgrim will not create the possibility of a new or different kind of accident from any accident previously evaluated because its removal does not degrade existing systems and because its function is enveloped by the turbine trip scram that remains in technical specifications.

The proposed change to the allowable maximum pressure setpoint results from a recalculation of maximum allowable scram bypass pressure that ensures Pilgrim is operated within the boundaries established to prevent or mitigate the effects of certain accident sequences described in the FSAR.

Hence, the proposed change supports the existing Pilgrim analysis and does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Pilgrim Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

Removal of LCVS will not increase the probability of occurrence or the consequences of a loss-of-vacuum transient because the low vacuum turbine trip scram provides sufficient protection to prevent plant damage and offsite consequences. The turbine trip is also a more direct variable for reactor protection. Therefore, LCVS is not a distinct element of Pilgrim's accident analysis and its removal does not impact Pilgrim's safety margin. Hence, removal of LCVS from technical specifications will not involve a significant reduction in the margin of safety.

The proposed amendment also maintains the margin of safety as defined by Pilgrim's safety analysis by changing the existing maximum allowable turbine first stage pressure permitting scram bypass from 305 psig to the more conservative 112 psig. The change is proposed because information supplied by Pilgrim's NSSS [nuclear steam system supplier]

[* * *] required recalculation of this setpoint to support the margin of safety under conditions and considerations discussed in the above item 1B1. Therefore, the proposed amendment 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: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

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

NRC Project Director: Walter R. Butler

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

Date of amendment request: September 27, 1993

Description of amendment request: The proposed amendment would modify Technical Specification (TS) section 6.3.1, Unit Staff Qualifications, to make that section consistent with the current requirements of Part 55 of Title 10 of the Code of Federal Regulations (10 CFR 55). The proposed amendment would also delete TS section 6.4.1, Training, because the requirements associated with training are now contained in 10 CFR 55 and 10 CFR 50.120.

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 Technical Specification changes are administrative changes to eliminate inconsistencies with the current regulations for unit staff qualifications and training programs. The proposed changes are being made to remove language describing or committing to any previous training programs, since the training programs at PNPP have been accredited and certified in accordance with the revised 10 CFR 55 and 10 CFR 50.120 rules, GL 87-07 and NUREG-1262. The proposed changes delete reference to the March 28, 1980, NRC letter (the Denton Letter) for licensed operator qualifications and training programs and, for licensed operator qualifications, will substitute compliance with the requirements of 10 CFR 55. The proposed changes also include the deletion of Specification 6.4.1 "Training," since training of both licensed operators and other appropriate unit staff personnel is now governed by regulations (10 CFR 55 and 10 CFR 50.120).

The proposed changes will have no significant adverse impact on accident probability or consequence. The NRC, during the rulemaking process, has considered any impact that licensed operator qualifications and training programs may have on accidents previously evaluated, and by promulgation of the revised 10 CFR 55 rule, concluded that this impact remains unchanged as long as licensed operator training programs are certified to be accredited and based on a systems approach to training in accordance with GL 87-07. CEI provided such certification for PNPP Unit 1 by letter PY-CEI/NRR-0866L dated June 9, 1988. The proposed Technical Specification changes take credit for the INPO accreditation of the licensed operator and other nuclear power plant personnel training programs, and continued compliance with the requirements of 10 CFR 55 and 10 CFR 50.120 is required regardless of any reference to them within the Technical Specifications. Therefore, the proposed changes do not increase the probability or consequences of an accident previously evaluated.

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

The proposed Technical Specification changes are administrative changes to eliminate inconsistencies with the current regulations for qualifications and training programs. The NRC, during the rulemaking process, has considered any impact that licensed operator qualifications and training programs may have on the possibility of a new or different kind of accident from any accident previously evaluated, and by promulgation of the revised 10 CFR 55 rule, concluded that this impact remains unchanged as long as licensed operator training programs are certified to be accredited and based on a systems approach to training in accordance with GL 87-07. CEI provided such certification for PNPP Unit 1 by letter PY-CEI/NRR-0866L dated June 9, 1988. The proposed Technical Specification changes take credit for the INPO accreditation of the licensed operator and other nuclear powerplant personnel training programs, and continued compliance with the requirements of 10 CFR 55 and 10 CFR 50.120 is required regardless of any reference to them within the Technical Specifications. Additionally, the proposed Technical

Specification changes do not affect plant design, hardware, system operation, or procedures. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed Technical Specification changes are administrative changes to eliminate inconsistencies with the current regulations for qualifications and training programs. The licensed operator qualifications and training programs will continue to be required to comply with the requirements of 10 CFR 55. The NRC, during the rulemaking process, has considered any impact that licensed operator qualifications and training programs may have on the margin of safety, and by promulgation of the revised 10 CFR 55 rule, concluded that this impact remains unchanged when licensees certify that their licensed operator training programs are accredited and based on a systems approach to training in accordance with GL 87-07. CEI provided such certification for PNPP Unit 1 by letter PY-CEI/NRR-0866L dated June 9, 1988. The NRC has concluded, as stated in NUREG-1262, that the standards and guidelines applied by INPO in their training accreditation program are equivalent to those put forth or endorsed by the NRC. As a result, maintaining INPO accredited systems based licensed operator training programs is equivalent to maintaining NRC approved licensed operator training programs which conform with applicable NRC RGs or NRC endorsed ANSI/ANS standards. The margin of safety is maintained by virtue of maintaining INPO accredited licensed operator and other nuclear power plant personnel training programs and through continued compliance with the requirements of 10 CFR 55 and 10 CFR 50.120. Therefore, the proposed changes do not reduce the margin of safety.

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

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

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

NRC Project Director: John N. Hannon

Commonwealth Edison Company, Docket Nos. 50-295 and 50-304, Zion Nuclear Power Station, Units 1 and 2, Lake County, Illinois

Date of amendment request: November 4, 1993

Description of amendment request: The proposed amendment would revise the Technical Specifications by changing the steam generator safety valve surveillance frequency and acceptance criteria.

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 of occurrence or consequences of an accident previously evaluated?

The probability of an accident previously evaluated has not been increased. The proposed change does not change the fundamental function or capability of the MSSVs [Main Steam Safety Valves] as described in the UFSAR [Updated Final Safety Analysis Report]. This change does not affect any initiators or precursors of an accident previously evaluated. This change will not increase the likelihood that a transient initiating event will occur because most transients are initiated by equipment malfunction and/or catastrophic system failure. Since the proposed change does not involve the introduction of new or redesigned plant equipment, these failure mechanisms are not impacted.

The consequences of accidents previously evaluated are not increased. The proposed change does not involve any equipment modifications which could adversely affect the expected accident sequence. Although the frequency of the MSSV surveillance testing is effected by the change, the frequency at which MSSV surveillances are performed is not assumed in any analyzed event. The changes in testing frequency are consistent with the ASME/ANSI Standard. The ASME/ANSI Standard has been applied extensively throughout the industry and demonstrated adequate by the resulting industry experience. Therefore, accident analyses assumptions reflected in the affected Surveillance Requirements will still be verified on a frequency sufficient to ensure that the assumptions are reliably maintained.

The role of these valves is in the mitigation of design basis accidents and transients. The effect of allowing the Zion station MSSV lift setpoint tolerance to increase from the currently required plus or minus 1 percent to the plus or minus 3 percent consistent with the ASME/ANSI Standard has been evaluated for all non-LOCA and LOCA design basis requirements. The plus or minus 3 percent tolerance for the MSSV setpoints was assumed in the VANTAGE5 Reload Transition Safety Report for the Zion Units 1 and 2. In all cases, either a reanalysis incorporating the increased MSSV setpoint tolerance continued to show results within acceptance limits, or the MSSV setpoints were determined not to affect the licensing basis results. Even though the plus or minus 3 percent tolerance has been shown to be acceptable, the proposed change conservatively requires the MSSV setpoints to be restored to within plus or minus 1 percent of the required value after testing. The remaining acceptance criteria of the IST program are at least as restrictive as existing Technical Specification requirements and ensure that an equivalent or greater degree of MSSV operational readiness is provided.

Additionally, the relocation of Surveillance details to the IST program and its implementing procedures will not increase the probability or consequences of a previously evaluated accident since adequate control of the requirements is provided by the 10 CFR 50.59 review process and ASME Section XI requirements incorporated by 10 CFR 50.55a(g). Therefore, this change does 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 previously analyzed?

The proposed change does not alter the design of the MSSVs or their function to protect against overpressure events. The proposed change does not introduce any new equipment, equipment modifications, or any new or different modes of plant operation. Therefore, the proposed change does not introduce any new failure modes and the plant will continue to be operated within acceptable limits. In addition, the proposed change still provides adequate assurance the MSSVs will be maintained operable.

For the reasons described above, there is no possibility that the proposed change creates a new or different kind of accident from any previously analyzed in the UFSAR.

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

The proposed change incorporates the industry standard testing requirements of Section XI of the ASME Code and applicable Addenda for the MSSVs. The Zion IST program requirements and implementing procedures have been developed in accordance with the ASME Section XI requirements to ensure component degradation is detected before the component is incapable of performing its intended safety function.

Although the frequency of the MSSV surveillance testing is affected by the change, the frequency at which MSSV surveillances are performed is not assumed in any analyzed event. The changes in testing frequency are consistent with the ASME/ANSI Standard. The ASME/ANSI Standard has been applied extensively throughout the industry and demonstrated adequate by the resulting industry experience. Any reduction in a margin of safety is insignificant since the extension of the surveillance intervals is justified based on accepted industry practice and compliance with ASME Section XI as mandated by 10 CFR 50.55(a)g. In addition, the proposed change has the potential to reduce testing that is typically done at power. Therefore, the proposed change reduces the risk of an unexpected plant transient that may be caused by online testing of the MSSVs.

The effect of allowing the Zion station MSSV lift setpoint tolerance to increase from the currently required plus or minus 1 percent to the plus or minus 3 percent consistent with the ASME/ANSI Standard has been evaluated for all non-LOCA and LOCA design basis requirements. The plus or minus 3 percent tolerance for the MSSV setpoints was assumed in the VANTAGE5 Reload Transition Safety Report for the Zion Units 1 and 2. In all cases, either a reanalysis incorporating the increased MSSV setpoint tolerance continued to show results within the acceptance limits, or the MSSV setpoints were determined not to affect the licensing basis results. Although the plus or minus 3 percent tolerance has been shown to be acceptable, the proposed change conservatively requires the MSSV setpoints to be restored to within plus or minus 1 percent of the required value after testing. Therefore, modifying the applicable Technical Specification Surveillance Requirements for the MSSVs in accordance with the industry standards will not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60690

NRC Project Director: James E. Dyer

Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: October 27, 1993

Description of amendment request: The proposed amendment would relocate the requirements in Technical Specification (TS) 3/4.3.3.2 regarding incore detectors from the TSs to the Safety Analysis Report.

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

Criterion 1 - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The proposed change relocates incore detection system requirements from the Technical Specification (TS) to the Safety Analysis Report (SAR) consistent with the Nuclear Regulatory Commission (NRC) Policy Statement on Technical Specification Improvements. The ANO-2 [Arkansas Nuclear One, Unit 2] incore detection system is not required for plant safety since it does not initiate any direct safety-related function during anticipated operational occurrences or postulated accidents. The primary function of the incore detectors is to provide inputs to the Core Operating Limits Supervisory System (COLSS) for monitoring of core parameters. The COLSS is independent of the plant protection system. The CPCs [Core Protection Calculators] operate independently of COLSS, using the excore detectors to preserve plant safety parameters. The proposed change does not affect any material condition of the plant that could directly contribute to causing or mitigating the effects of an accident. The TS will continue to define the Limiting Conditions of Operation required to ensure that reactor core conditions during operations remain within the initial conditions assumed in the SAR. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

Because the proposed change does not change the design, configuration, or method of operation of the plant, it does not create the possibility of a new or different kind of accident. The incore detection system is not a part of plant control instruments or engineered safety feature actuation circuits. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.

This change does not decrease the margin of safety since the incore detection system is not required for plant safety. The system does not initiate any direct safety-related function during anticipated operational occurrences or postulated accidents. The proposed change relocates the incore detection system requirements from the TS to the SAR. Changes to the SAR are controlled under the criteria specified by 10CFR50.59. The proposed change will have no adverse impact on the plant protection system nor will any protective boundary or safety limit be affected. Therefore, this 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 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: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801

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

NRC Project Director: William D. Beckner

Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: October 27, 1993

Description of amendment request: The proposed amendment would relocate the requirement to verify the correct position of each electrical and/or mechanical position stop for the Emergency Core Cooling System throttle valves within 4 hours of each valve stroking operation or maintenance on the valve, to procedures that control the maintenance and operation of these valves.

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

Criterion 1 - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The proposed change relocates the requirements concerning verification of correct position stop position [i.e. verification of the correct position of each position stop] following maintenance to licensee controlled documents, consistent with NUREG 1432 "Improved Standard Technical Specifications for Combustion Engineering Plants." The Operations and Maintenance procedures, which will contain these requirements, are controlled under the criteria set forth in 10CFR50.59. The position of the position stops will be verified following maintenance or adjustment of the ECCS [emergency core cooling system] throttle valves and periodically thereafter. The position stops will be verified at least every 18 months, as requiredby TS [Technical Specification] 4.5.2.g.2. The relocation of these position stop verification requirements is considered to be administrative in nature.

The ECCS throttle valves are not initiators of any accident previously evaluated. Therefore, the deletion of the requirement to verify the correct position of the position stops within 4 hours following completion of each valve stroking operation will not result in the increase in the probability of any accident previously evaluated. The ANO-2 [Arkansas Nuclear One, Unit 2] maintenance history reviewed for the eight ECCS throttle valves subject to the requirements of TS 4.5.2.g.1 has shown only four documented instances of failure of the open position to stop valve travel at the correct position since the beginning of 1985.

The deletion of the requirement to verify the correct position of the position stops following completion of each valve stroking operation will result in fewer challenges to the proper operation of the ECCS throttle valves. The probability of inducing a position stop failure due to valve stroking operations is considered to be highly unlikely. The process of position stop setting verification results in unnecessary additional challenges that could result in overall lower valve reliability. Therefore, position stop setting verification beyond that required for post-maintenance testing and periodically thereafter, as required by TS 4.5.2.g.2, is considered unwarranted. Since valve reliability will not be decreased as a result of this change, there is no significant increase in the consequences of any accident previously analyzed.

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

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

Because the proposed change does not change the design, configuration, or method of operation of the plant, it does not create the possibility of a new or different kind of accident. The proposed change does not allow the ECCS throttle valves to be operated in any new or different way from what is currently allowed.

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

Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.

The proposed change relocates the requirements concerning verification of correct position stop position following maintenance to licensee controlled documents, consistent with NUREG 1432 "Improved Standard Technical Specifications for Combustion Engineering Plants." The Operations and Maintenance procedures, which will contain these requirements, are controlled under the criteria set forth in 10CFR50.59. The position of the position stops will be verified following maintenance or adjustment of the ECCS throttle valves and periodically thereafter. The position stops will be verified at least every 18 months, as required by TS 4.5.2.g.2. The relocation of these position stop verification requirements is considered to be administrative in nature and does not involve a significant reduction in the margin of safety.

The ANO-2 maintenance history reviewed for the eight ECCS throttle valves subject to the requirements of TS 4.5.2.g.1 has shown only four documented instances of failure of the open position stop to stop valve travel at the correct position since the beginning of 1985. The deletion of the requirement to verify the correct position of the position stops following completion of each valve stroking operation will result in fewer challenges to the proper operation of the ECCS throttle valves. The probability of inducing a position stop failure due to valve stroking operations is considered to be highly unlikely. The process of position stop setting verification results in unnecessary additional challenges that could result in overall lower valve reliability. Therefore, position stop setting verification beyond that required for post-maintenance testing and periodically thereafter, as required by TS 4.5.2.g.2, is considered unwarranted. Since valve reliability will not be decreased as a result of this change, there is no significant reduction in the margin of safety.

Therefore, this 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 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: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801

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

NRC Project Director: William D. Beckner

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

Date of amendment request: October 22, 1993

Description of amendment request: The proposed changes would amend Technical Specifications (TSs) by modifying the testing frequencies for the drywell bypass test and airlock test, relocating certain drywell airlock tests from the TSs to administrative procedures, and incorporating various improvements from the Improved Standard Technical Specifications (NUREG-1434, Revision 0).

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

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

1. The changes to Technical Specification 1.10 are purely administrative since the intent is to make the numbering consistent with the other proposed Technical Specifications. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The relocation of drywell leakage rate requirements of LCO [limiting condition for operation] 3.6.2.2 as a supporting surveillance for TS 3/4.6.2.1 (DRYWELL INTEGRITY) is only an administrative presentation change consistent with the guidance of NUREG-1434, Standard Technical Specifications, General Electric Plants, BWR/6 (Ref. 3). Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

3. The proposed change relocates certain details from the GGNS [Grand Gulf Nuclear Station] Technical Specifications (TS) to the TS Bases, UFSAR [updated final safety analysis report] or procedures. The TS Bases, UFSAR and procedures containing the relocated information will be maintained in accordance with 10 CFR 50.59 and are subject to the change control provisions in the Administrative controls section of Technical Specifications. Since any changes to the TS Bases, UFSAR or procedures will be evaluated per the requirements of 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

4. The proposed change relocates certain details from GGNS TS to the TS Bases, UFSAR or procedures. The TS Bases, UFSAR and procedures containing the relocated information will be maintained in accordance with 10 CFR 50.59 and are subject to the change control provisions in the Administrative Controls section of TS. Since any changes to the TS Bases, UFSAR or procedures will be evaluated per the requirements of 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

5. This change would delete the restriction which prevents use of the generic surveillance extension allowance for drywell bypass leakage testing. Drywell bypass leakage is not considered as the initiator for any previously evaluated accidents and, therefore, revising the surveillance frequency will not significantly increase the probability of any previously evaluated accident. Further, since the change maintains testing to verify the analyzed bypass leakage is not exceeded following an accident and does not result in any change in the response of the drywell to an accident, the change does not increase consequences of any accident previously evaluated.

6. The proposed change deletes an administrative requirement to obtain NRC staff review and approval of the test schedule for drywell bypass leakage tests, if one test fails to meet the specified limit. Test schedules are not used as the initiator of any accident. Therefore, the probability of any accident previously evaluated is not increased. This proposed deletion does not change the requirement for limiting drywell bypass leakage, only the requirement to receive NRC staff review and approval of a schedule for doing the test. Therefore, the consequences of previously evaluated accidents are not increased.

The proposed change in frequency for the drywell bypass leakage surveillance will continue to ensure that no paths exist through passive drywell boundary components to permit gross leakage from the drywell to the primary containment air space and result in bypassing the containment pressure-suppression feature beyond the design basis limit. The GGNS Mark III containment system satisfies General Design Criterion 16 of Appendix A to 10 CFR Part 50. Maximum drywell bypass leakage was determined previously by reviewing the full range of postulated primary system break sizes. The limiting case was a primary system small break LOCA and yielded a design allowable drywell bypass leakage rate limit of 35,000 scfm [standard cubic foot/feet per minute]. The TS acceptable limit for the bypass leakage surveillance is 10% (i.e., 3,500 scfm) of this design basis value. The design basis drywell bypass leakage limit will not be affected by these proposed changes. Drywell integrity has been reliable at GGNS as indicated by past surveillances. The most recent bypass leakage value was approximately 1.8% of the design allowable leakage rate limit. GGNS is committed to maintaining programmatic and oversight controls that ensure that drywell bypass leakage remains a small fraction of the design allowable leakage limit. Therefore, the proposed changes do not significantly increase the consequences of an accident previously evaluated.

In order to analyze the impact of this proposal, the probability of excessive drywell bypass leakage is very conservatively assumed to be 1E-2 per year. A small LOCA initiator has a frequency of occurrence of 1E-3 per year in the GGNS IPE [individual plant examination]. The containment spray system was modeled in the GGNS IPE and has a failure probability to function on demand of approximately 1E-2 per year for a LOCA initiator given a core damage accident. The resultant frequency for an overpressure failure of containment due to excessive dr;well leakage is conservatively estimated to be less than 1E-7 per year. This is a very low frequency event, and is on the order of the low frequency severe accident events considered in the GGNS IPE.

Since the resulting potential release would be much smaller than in a severe accident sequence of comparable frequency, it is clearly bounded by the GGNS IPE analysis results. This sequence would not increase overall plant risk. Therefore, the proposed changes do not have any significant risk impact to accidents previously evaluated.

In the unlikely event of a design basis accident, primary containment should maintain its integrity as designed since the margin of safety is not reduced. Secondary containment integrity, in conjunction with the standby gas treatment system (SGTS) with redundant 100% capacity trains, would also mitigate the consequences of a design basis accident. SGTS is an engineered safety feature and is described in GGNS UFSAR Section 6.5.3.

7. The proposed change would allow continued operation with an inoperable drywell airlock door interlock mechanism. Having both drywell airlock doors open at the same time is not an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the frequency of such accidents. The proposed change provides actions with appropriate compensatory measures to maintain a level of safety equivalent to compliance with the LCO. These actions do not result in airlock function different than assumed in any accident. Therefore, this change does not significantly increase the consequences of any previously analyzed accident.

The proposed change would allow the temporary opening of the remaining OPERABLE door for the purpose of making repairs to a drywell airlock door and for a limited period of time for purposes other than making repairs. This change does not affect the airlock design or function, and failure of an airlock is not identified as the initiator of any event. Therefore, this proposed change does not involve an increase in the probability of an accident previously evaluated. The change to allow the temporary opening of the one OPERABLE door for the purpose of making repairs results in a potential increase in consequences should an accident occur while it is open, but this increase is minimized through administrative controls and offset by the avoided potential consequences of a transient during shutdown. The potential for increased consequences resulting from the combination of: (1) the frequency of experiencing an inoperable airlock door such that the temporary opening of the OPERABLE door is required for access to repair; (2) the brief period that the OPERABLE door would be opened for access (typically on the order of one minute per entry/exit); (3) the proximity of an individual to accomplish closure; and (4) the occurrence of an event of sufficient magnitude to cause an immediate containment pressure increase such that an airlock door could not be closed; is not considered to be significant. Additionally, providing the ability to eliminate the potential consequences of: (1) extended operation with only one OPERABLE door closed (not allowing repairs to be made to restore the second door to OPERABLE status); and (2) the transient of plant shutdown to follow (due to inability to perform the overall airlock test); further minimizes the consequences. The allowance is proposed have strict administrative control which will provide assurance that any associated potential consequences are minimized. Finally, the allowed time for both doors to be open is not expected to exceed the currently allowed time for required action when drywell integrity is determined to not be met. Therefore, these proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

This change would delete the restriction which prevents use of the generic surveillance extension allowance for drywell airlock leakage testing. Drywell airlock leakage is not considered as the initiator for any previously evaluated accidents and, therefore, revising the surveillance frequency will not significantly increase the probability of any previously evaluated accident. Further, since the change maintains testing to verify that the analyzed airlock leakage is not exceeded following an accident and the proposed change does not alter the response of the drywell to an accident, the change does not increase the consequences of any previously analyzed accident.

This change may increase the surveillance time interval of the drywell airlock leakage test. The current specification requires that it be conducted at each COLD SHUTDOWN if not conducted in the previous 6 months. If no shutdowns occur between refuelings, the time interval is the same as proposed. Therefore, there is no substantial change to the time interval. Further, there is no effect from a shutdown that would cause the airlock capabilities to be reduced. Therefore, this proposed change does not involve an increase in the probability of an accident previously evaluated. Further, since the change impacts only the frequency of verification and does not alter the response of the equipment to an accident, the change does not increase the consequences of any previously analyzed accident.

This change would increase the surveillance time interval of the drywell airlock door interlock so that it is not required to be performed unless the drywell airlock doors are to be opened for drywell entry. The proposed change does not affect the drywell airlock design or function. Additionally, a failure of an airlock is not identified as the initiator of any event. Therefore, this proposed change does not involve an increase in the probability of an accident previously evaluated. Further, since the change impacts only the frequency of verification and does not result in any change in the response of the equipment to and accident, the change does not increase the consequences of any previously analyzed accidents.

8. Calculations show that the maximum possible leakage possible with failed drywell airlock seals would not compromise the drywell safety function. The proposed change does not affect the drywell airlock design or function. Additionally, a failure of an airlock is not identified as the initiator of any event. The UFSAR containing the relocated information is maintained in accordance with 10 CFR 50.59 and is subject to the change control provisions in the Administrative Controls section of Technical Specifications. Since any changes to the UFSAR will be evaluated per the requirements of 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. Therefore, relocation of the airlock seal OPERABILITY requirements to the UFSAR does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

1. The proposed changes to Technical Specification 1.10 are purely administrative since the intent is to make the numbering consistent with the other proposed Technical Specifications. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

2. The proposed relocation of drywell leakage rate requirements of LCO 3.6.2.2 as a supporting surveillance for TS 3/4.6.2.1 (DRYWELL INTEGRITY) is only an administrative presentation change consistent with the guidance of NUREG-1434 (Ref. 3). Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed relocation of requirements does not involve a physical alteration of the plant (no new or different type of equipment will be installed) nor does it change the

methods governing normal plant operation. The proposed change will not impose or eliminate any requirements. Adequate control of the information will be maintained in the UFSAR. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

4. The proposed relocation of requirements does not involve a physical alteration of the plant (no new or different type of equipment will be installed) nor does it change the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements. Adequate control of the information will be maintained in the UFSAR. Thus, the change proposed does not create the possibility of a new or different kind of accident from any accident previously evaluated.

5. The proposed deletion does not alter equipment design, equipment capabilities, or operation of the plant. Further, since the change impacts only the test frequency for verification of leaktightness and does not result in any change in the response of the equipment to an accident, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

6. The proposed change modifies the surveillance frequency for drywell bypass leakage and deletes an administrative requirement to get NRC staff review and approval of the test schedule. The change does not alter equipment design or capabilities. The changes do not present any new or additional failure mechanisms. The drywell is passive in nature and the surveillance will continue to verify that its integrity has not deteriorated. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

7. The proposed change does not alter equipment design or capabilities, but do allow operation of the plant with equipment that is incapable of performing its safety function. However, the change does include compensatory measures which will maintain a level of safety equivalent to the capabilities of the equipment. Drywell airlocks are designed and assumed to be used for entry and exit. Their operation does not interface with the reactor coolant system or any controls which could impact the reactor coolant pressure boundary or its support systems. The change impacts the test frequency for verification of airlock leaktightness and does not result in any change in the response of the equipment to an accident. Furthermore, brief periods of loss of drywell integrity are acknowledged in the existing license; TS 3.6.2.1 allows 1 hour to restore loss of drywell integrity prior to requiring a plant shutdown. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

8. The proposed relocation of requirements does not affect the drywell airlock design or function. Calculations show that the maximum leakage possible with failed drywell airlock seals would not compromise the drywell safety function. Additionally, failure of an airlock is not identified as the initiator of any event. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

1. The changes to Technical Specification 1.10 are purely administrative since the intent is to make the numbering consistent with the other proposed Technical Specifications. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

2. The relocation of drywell leakage rate requirements of LCO 3.6.2.2 as a supporting surveillance for TS 3/4.6.2.1 (DRYWELL INTEGRITY) is only an administrative presentation change consistent with the guidance of NUREG-1434 (Ref. 3). Therefore, this change does not involve a significant reduction in a margin of safety.

3. The relocation of requirements will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the TS to the TS Bases, UFSAR or procedures are the same as the existing Technical Specifications. Since any future changes to these requirements in the TS Bases, UFSAR or procedures will be evaluated per the requirements of 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed. Also, since the proposed change is consistent with NUREG-1434 (Ref. 3) as approved by the NRC Staff, revising the TS to reflect the approved level of detail ensures no significant reduction in the margin of safety.

4. The relocation of requirements will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the TS to the TS Bases, UFSAR or procedures are the same as the existing Technical Specifications. Since any future changes to these requirements in the TS Bases, UFSAR or procedures will be evaluated per the requirements of 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed. Also, since the proposed change is consistent with NUREG-1434 (Ref. 3) as approved by the NRC Staff, revising the TS to reflect the approved level of detail ensures no significant reduction in the margin of safety.

5. The proposed deletion impacts only the test frequency to be used for verification of the drywell bypass leakage. The limits on the allowable leakage are not revised and must continue to be met. Therefore, the change does not involve a significant reduction in the margin of safety.

6. The proposed change modifies the surveillance frequency for drywell bypass leakage and deletes an administrative requirement to get NRC staff review and approval of the test schedule. Reliability of drywell integrity is evidenced by the measured leakage rate during past drywell bypass leakage surveillances. Appropriate design basis assumptions will be upheld, even when combined with the complementary bypass leakage surveillances as proposed. The surveillance acceptance leakage rate is 10% of the design allowable drywell bypass leakage limit of 35,000 scfm. Margins of safety would not be reduced unless leakage rates exceeded the design allowable drywell bypass leakage limit. Therefore, the proposed change does not reduce the margin of safety.

7. This change permits the use of dedicated personnel to provide compensatory actions in place of automatic equipment for a limited time. These administrative controls continue to provide an adequate drywell boundary should an accident occur. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

The design, function, and OPERABILITY requirements for the drywell airlock remains unchanged with this proposed revision. Drywell leak rate limits are unaffected. The proposed change to allow the temporary opening of the one OPERABLE door for the purpose of repairing an inoperable airlock door and for purposes other than repairing an inoperable airlock door (for a limited time), is not considered to be a significant reduction in the margin of safety. The combination of: (1) the frequency of experiencing an inoperable airlock door such that drywell entry is required for access to repair; (2) the brief period the OPERABLE door would be opened for access (typically on the order of one minute per entry/exit); (3) the proximity of a dedicated individual to accomplish closure; and (4) the occurrence of an event of sufficient magnitude to cause an immediate containment pressure increase such that an airlock door could not be closed; are not considered to be representative of a significant reduction in the margin of safety. Additionally, providing the ability to eliminate any reduction in safety resulting from the combination of: (1) extended operation with only one OPERABLE door closed (not allowing repairs to be made to restore the second door to OPERABLE status); and (2) the transient of plant shutdown to follow (due to inability to perform the overall airlock test); further minimizes any reduction in the margin of safety. The allowance is proposed have strict administrative control which will provide assurance that any associated safety reduction is further minimized. Finally, the allowed time for both airlock doors to be open is not expected to exceed the currently allowed time for required action when drywell integrity is determined to not be met. Therefore, the proposed changes do not reduce the margin of safety.

The proposed change affecting frequency of testing impacts only the verification of drywell airlock leakage. The limits on the allowable leakage are not revised and must continue to be met. The changes in testing frequency will not reduce the reliability of the drywell airlock hardware. The surveillances will continue to provide sufficient assurance of OPERABILITY. Therefore, the proposed changes do not reduce the margin of safety.

8. The proposed change does not adversely affect design or performance of the drywell or primary containment safety functions.

Drywell integrity will continue to be surveilled by means of the proposed periodic drywell bypass leakage test, performance of the drywell airlock door latching and interlock mechanism surveillance, and performance of additional surveillances including drywell isolation valves. The combination of these surveillances will provide adequate assurance that drywell bypass leakage will not exceed the design basis limit. Evaluation of bypass leakage values for complete failure of the drywell airlock door seals determined that the drywell airlock door seals are not required to maintain the design basis assumption for limited drywell bypass leakage. Therefore, the proposed change does not reduce a margin of safety.

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

Local Public Document Room location: Judge George W. Armstrong Library, Post Office Box 1406, S. Commerce at Washington, Natchez, Mississippi 39120

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

NRC Project Director: William D. Beckner

Illinois Power Company and Soyland Power Cooperative, Inc., Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Date of amendment request: November 4, 1993

Description of amendment request: The proposed amendment would modify Technical Specification 3/4.8.1.1, "AC Sources-Operating," by removing Surveillance Requirement 4.8.1.1.2.e.1 from the technical specifications and relocating it to plant controlled programs. This surveillance requirement subjects each diesel generator to an inspection in accordance with the manufacturer's recommendations. The proposed action is consistent with the improved Standard Technical Specifications for BWR/6 facilities (NUREG-1434).

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 is consistent with the improved Standard Technical Specification (NUREG-1434) and does not result in any changes to the existing plant design. The diesel generators will continue to be inspected in accordance with the manufacturer's recommendations as part of the Clinton Power Station preventive maintenance program. Since the change does not impact the ability of the diesel generators and the AC electrical power sources to perform their function, this change does not result in a significant increase in the consequences of any accident previously evaluated. The diesel generators will continue to function as designed and will continue to be tested as previously tested. Therefore, the proposed change will not impact the probability of occurrence of any accident previously evaluated.

(2) This request does not result in any change to the plant design nor does it involve a significant change in current plant operation. The diesel generators will continue to be inspected as recommended by the manufacturer and the remaining surveillance requirements will not be changed. The change merely permits taking credit for current preventive maintenance activities without specifically requiring the inspection activity in the Technical Specifications. As a result, no new failure modes will be introduced, and the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) The proposed request does not adversely impact the reliability of the diesel generators. As stated above, the manufacturer's recommended inspections will continue to be performed. In addition, the diesel generators will continue to perform their design functions. This request does not involve an adverse impact on diesel generator operation or reliability. Since the diesel generator function is not affected by the proposed change, this request 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: Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727

Attorney for licensee: Sheldon Zabel, Esq., Schiff, Hardin and Waite, 7200 Sears Tower, 233 Wacker Drive, Chicago, Illinois 60606

NRC Project Director: James E. Dyer

Illinois Power Company and Soyland Power Cooperative, Inc., Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Date of amendment request: November 4, 1993

Description of amendment request: The proposed amendment would modify Technical Specification 3/4.8.2.1, "DC Sources-Operating," by deleting the requirement that the plant be shut down to perform the required battery capacity or service testing.

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 allows removal of the Division III or IV battery from service for testing during plant operation. Because of a two-hour requirement for restoration, it is not expected that Division I or II batteries would be removed from service during plant operation. Removal of any DC subsystem from service does not render any other subsystem inoperable. Clinton Power Station Updated Safety Analysis Report Section 8.3.2 states that the system design allows for the single failure or loss of any redundant DC subsystem during simultaneous accident and loss of offsite power conditions without adversely affecting safe shutdown of the plant. Since all required functions for safe shut down of the facility in response to an accident can be performed by the Division I and II DC subsystems, permitting the Division III or IV 125 VDC subsystem to be out of service during plant operation would not result in an increase in the consequences of any accidents previously evaluated. Loss of the DC Electrical Distribution System is not itself an initiator of any previously evaluated accident. The proposed change would therefore have no impact on the probability of occurrence of an accident previously analyzed.

(2) This request does not result in any change to the plant design nor does it involve a change in current plant operation. The proposed change would have no effect on the way the battery capacity test is performed. Maintenance on the Division III or IV battery would increase reliability of the affected battery, and post-modification testing would ensure the battery is operable in accordance with the vendor recommendations prior to being returned to service. As a result, no new failure modes would be introduced, and the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) The proposed request does not adversely impact the reliability of the DC Electrical Distribution System. The remaining three divisions of DC power would continue to perform the system's design function while the Division III or IV battery is inoperable for testing. Further, the Technical Specifications permit the Division III and/or IV batteries, as well as the High Pressure Core Spray system itself, to be inoperable for limited periods of time during reactor operation. Since the proposed change would not adversely impact system operation or reliability, and since the DC Electrical Distribution System function would not be adversely affected by the proposed change, this request 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: Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727

Attorney for licensee: Sheldon Zabel, Esq., Schiff, Hardin and Waite, 7200 Sears Tower, 233 Wacker Drive, Chicago, Illinois 60606

NRC Project Director: James E. Dyer

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

Date of amendment request: November 3, 1993

Description of amendment request: The proposed amendment would modify License Condition 2.C.(4) and delete Technical Specification (TS) 3/4.3.8, "Turbine Overspeed Protection System." License Condition 2.C.(4) required the licensee to submit for NRC approval a turbine system maintenance program based on the manufacturer's calculations of missile generation probabilities. The proposed change to License Condition 2.C.(4) would indicate that this requirement has been satisfied. The deletion of TS 3/4.3.8 would provide the licensee with the flexibility to implement the manufacturer's recommendations for turbine steam valve surveillance test requirements. The turbine steam valve surveillance test requirements based on manufacturer's recommendations would be contained in the Updated Safety Analysis Report.

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 operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

With the approval of this Amendment, preventative maintenance, testing, and inspections of the Turbine Overspeed Protection System will remain governed by an approved turbine system maintenance program, described in the USAR [Updated Safety Analysis Report]. To maintain turbine system reliability, controlled procedures are in place implementing manufacturer's recommendations. In evaluating the turbine system maintenance program (NRC approved by letter dated March 15, 1990 which satisfied License Condition 2.C.(4)) the Staff found the overall probability of generating a turbine missile at Nine Mile Point Unit 2 to be less than one in ten thousand (<1E-4) events per year. This probability, when combined with a 1E-3 probability (NUREG 1048, Supplement 6, Appendix U) for missile impact and essential system damage, yields an overall probability of less than one in ten million (<1E-7) events per year. Less than one in ten million (<1E-7) events per year is an acceptably low probability according to the criteria of NUREG 0800 and agrees with the initial staff finding in NUREG 1048. Consequently, the probability of a previously evaluated turbine missile accident will not increase.

The purpose of the Turbine Overspeed Protection System is prevention of an overspeed event, the precursor to a potential turbine fragment missile. Since the purpose of this system is preventative, it serves no function to mitigate any accident previously evaluated and thus does not affect the consequences of any analyzed accident.

Updating License Condition 2.C.(4) is administrative in nature and does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Accidents which include rapid Turbine Stop Valve closure as a response to some initiating event are not relevant to this discussion since in those instances the valves respond as designed.

The relevant accident resulting from a failure of the Turbine Overspeed Protection System is a turbine fragment missile as evaluated in Section 3.5.1.3 of the Nine Mile Point Unit 2 Updated Safety Analysis Report. Approval of this amendment would not change the operational characteristics of surveillance tests and would impose no new testing requirements, but rather relocate testing requirements from Technical Specifications to the USAR. Updating License Condition 2.C.(4) is administrative in nature and does not alter intent of any requirements. Therefore, approval of this amendment to delete Specification 3/4.3.8 and to update the License Condition 2.C.(4) signifying NRC approval would not create the possibility of a new or different kind of accident from the turbine missile accident previously evaluated.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

With the approval of this Amendment, Niagara Mohawk remains committed to the manufacturer's turbine reliability program. This turbine reliability program calculates the same maximum permissible probability for generation of a turbine missile as previously evaluated. This turbine missile generation probability, when combined with a favorable turbine orientation, results in the same, acceptably low, overall probability of turbine missile damage to essential systems and does not involve a reduction in the margin of safety.

Further, the approval of this Amendment will allow Niagara Mohawk to optimize the performance of testing and inspections in accordance with the manufacturer's recommendations and operational experience. Implementing the manufacturer's recommendations may lead to a reduced frequency of certain steam valve surveillance tests and a corresponding reduced probability of challenges to plant equipment and personnel, thereby enhancing the margin of safety. Updating License Condition 2.C.(4) is administrative in nature and does not alter intent of any requirements.

The deletion of Technical Specification 3/4.3.8 and associated bases and an update signifying satisfaction of the License Condition 2.C.(4) will not, therefore, decrease 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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005-3502.

NRC Project Director: Robert A. Capra

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

Date of amendment request: October 15, 1993

Description of amendment request: The proposed changes to Tables 3.8-1 and 3.8-2 would provide a maximum duration for which the radioactive effluent monitoring instrumentation may be out-of-service for the purpose of maintenance, performance of required tests, checks, calibrations, or sampling before the applicable action statement is entered. Additionally, (1) "sampling" is proposed to be added to the applicability statements within Tables 3.8-1 and 3.8-2 as an additional reason for the radioactive effluent monitoring instrumentation to be out-of-service and (2) the sentence "Auxiliary sampling must be initiated within 12 hours of initiation of this action statement" is proposed to be added to Action Statement D for Table 3.8-2.

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

NNECO [Northeast Nuclear Energy Company] has reviewed the proposed changes in accordance with 10CFR50.92 and has concluded that they do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

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

These changes address the operability requirements for radioactive effluent monitoring instrumentation outlined in Tables 3.8-1 and 3.8-2, and Action Statement D associated with Table 3.8-2 on page 3/4 8-8. The addition of a 12-hour channel inoperability time limit to the applicability statements within Tables 3.8-1 and 3.8-2 provides a specific duration for which radioactive effluent monitoring instrumentation may be out-of-service for the purpose of maintenance and performance of required tests, checks, calibrations, and sampling without entering the associated action statement. The 12-hour time limit was deemed appropriate based on previous historical performance of the maintenance on this instrumentation. The inclusion of sampling to the activities which may be performed during instrument service interruption is necessary to more accurately reflect routine work currently performed on these instruments. The addition of the sentence, "Auxiliary sampling must be initiated within 12 hours of initiation of the action statement" on page 3/4 8-8 provides specific guidance for periods of instrument inoperability beyond that specified in the applicability statements for iodine and particulate samplers. Auxiliary sampling for the Iodine and Particulate Monitoring Instrumentation requires setup of temporary monitoring equipment. As such, the 12-hour time allotment is appropriate for this action statement.

These changes provide clarification of the actions to be taken during instrument inoperability. The radioactive effluent monitoring instrumentation is passive and therefore does not affect design basis accident scenarios. These changes do not involve any alterations to plant equipment or procedures which would affect any operational modes or accident precursors. Therefore, the changes have no effect on the probability of occurrence of previously evaluated accidents, and have no effect on the consequences of previously evaluated accidents.

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

The changes described above do not involve physical modifications to the radioactive effluent monitoring instrumentation and, therefore, do not affect plant or operator response to an accident. T" e changes clarify operability requirements .ssociated with this instrumentation, which is pa..ive and, therefore, cannot initiate or mitigate any type of accident. The instrumentation serves to provide radiological information to the plant operator. As such, the proposed changes have no impact on design basis accidents, and the changes will not modify plant response or create a new or unanalyzed event. No new failure modes are introduced.

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

These changes provide specific operability requirements for radioactive effluent monitoring instrumentation and do not have any impact on the protective boundaries and, therefore, have no impact on the safety limits for these boundaries. The instrumentation associated with these changes does not provide a safety function and only serves to provide radiological information to plant operators. The instrumentation has no affect on the operation of any safety-related equipment. No hardware, software, or setpoint changes are involved in this wording change. These changes provide more definitive operability and surveillance requirements for radioactive effluent monitoring instruments. As such, these changes have no impact on the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that th . 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, Thames Valley State Technical College, 574 New London Turnpike, Norwich, Connecticut 06360.

Attorney for licensee: Gerald Garfield, Esquire, Day, Berry & Howard, Counselors at Law, City Place, Hartford, Connecticut 06103-34

NRC Project Director: John F. Stolz

Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Dockets Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: October 5, 1993

Description of amendment request: The amendment would revise the Plant Operating Review Committee (PORC) review, the Nuclear Review Board review, Radiological Environmental Monitoring Program requirements, position titles, and the organization chart in Appendix B consistent with Appendix A.

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 because they do not affect operation, equipment, or a safety related activity and are hence administrative in nature. Thus, these administrative changes cannot affect the probability or consequences of any accident.

2. The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated because these changes are purely administrative and do not affect the plant. Therefore, these changes cannot create the possibility of any accident.

3. The proposed changes do not involve a significant reduction in a margin of safety because the changes do not affect any safety related activity or equipment. These change are purely administrative in nature and do not affect the margin of safety.

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

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

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: Larry E. Nicholson, Acting

Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Dockets Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: October 27, 1993

Description of amendment request: The licensee proposes to change the Technical Specifications to 1) require the Senior Manager-Operations to hold a Senior Reactor Operator (SRO) license; and 2) delete the requirement for the a) Plant Manager or Superintendent-Operations, b) the Assistant Superintendent-Operations, and c) the Superintendent-Technical or the Engineer-Systems to hold an SRO license.

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

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

The probability of occurrence of an accident is based in part on the training and qualification requirements applicable to the personnel filling key plant management positions. Accordingly, the qualifications and scope of responsibilities applicable to plant management positions relative to the guidance in ANSI N18.1-1971, as described in Updated Final Safety Analysis Report (UFSAR) Section 13.2, "Organizational Structure," were originally reviewed and approved by the NRC during the initial plant licensing. Specifically, UFSAR Section 13.2.3, "Qualifications of Nuclear Plant Personnel," details the following correlation between plant management positions and the criteria in ANSI N18.1-1971:

[. . . See licensee's table in application]

Section 4.2.1, 'Plant Managers,' of ANSI N18.1-1971 states in part that '... The plant manager shall have acquired the experience and training normally required for examination by the AEC for a Senior Reactor Operator's License...' unless the plant organization includes one or more persons who are designated as principal alternates for the plant manager and who meet the nuclear power plant experience and training requirements established for the plant manager. The Plant Manager can conform to the criterion of ANSI N18.1-1971 Section 4.2.1 without holding an SRO License by acquiring nuclear plant experience and training. The Senior Manager-Operations is designated as a principal alternate to the Plant Manager. ANSI N18.1-1971, Section 4.2.2, 'Operations Manager,' states in part that at the time of '...appointment to the active position...the operations manager shall hold a Senior Reactor Operator's License.' Requiring the Senior Manager-Operations to hold an SRO License will continue to ensure conformance with this criterion. ANSI N18.1-1971, Section 4.3.2, 'Supervisors Not Requiring AEC Licenses,' does not include any recommendation that these managers have the training to be eligible for, or hold, an SRO license. ANSI N18.1-1971, Section 4.2.4, 'Technical Manager,' does not include any recommendation that the Technical Manager have the training to be eligible for, or hold, an SRO License.

The proposed TS change would continue to require that the individual responsible for the management of plant operations as well as day-to-day operating activities and conformance to the operating license, TS, and operating procedures demonstrate detailed operating knowledge and successfully complete training required to obtain and hold an SRO License, while deleting the unnecessary requirement that the Plant Manager or the Assistant Superintendent-Operations or the Superintendent-Technical or the Engineer Systems hold an SRO License. Also, licensed plant shift operators will continue to report to a management position filled by an individual who holds an SRO License.

Operations management and Technical management personnel would continue to maintain cognizance of pertinent plant, procedure, and TS changes by virtue of the responsibilities of their plant management positions, TS required PORC membership, and roles in the Emergency Response Organization. These responsibilities include review and or approval of proposed new or revised operating procedures and oversight of LOR training. Therefore, the qualifications of the Operations and Technical Management personnel will remain at the currently required level. Furthermore, these key plant management individuals who will no longer [be] required to hold an SRO License will be able to devote the time now spent in LOR training to increase their overview and involvement in plant operation and planning activities. Accordingly, the probability of occurrence of an accident previously evaluated in the Safety Analysis Report (SAR) that was based on the training and qualification of key plant management personnel is not increased by the proposed change to the current SRO License requirements.

The consequences of an accident previously evaluated in the SAR could be affected by the qualification of plant management personnel to which the plant operators report via the chain of command. As explained above, the proposed TS change to require the manager in the licensed operator chain of command to hold an SRO License will continue to meet the guidance provided by the applicable criteria in ANSI N18.1-1971.

This proposed change does not involve any changes to plant SSC, or in the manner in which plant SSC [structures, systems or components] are operated, maintained, modified, tested, or inspected. Therefore, the proposed TS change does not increase the consequences of accidents previously evaluated in the SAR.

Accordingly, as explained above, the proposed TS change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

This proposed change involving the qualification (e.g., obtain and hold [an] SRO License) of key plant management personnel cannot create the possibility of a new or different type of accident than previously evaluated in the SAR because no substantive change to the current requirements is involved as discussed above. Also, because the proposed TS change does not involve physical changes to plant SSC, the possibility of creating a different type of accident than previously evaluated in the SAR cannot be created. Therefore, the possibility of a different type of accident than previously evaluated in the SAR is not created.

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

The margin of safety of overall plant operating activities is based in part on the TS requirements that personnel serving in key plant management positions satisfy qualification criteria specified in ANSI N18.1-1971. The proposed change to the TS does not reduce these established qualifications that key plant management personnel must currently satisfy. In addition, implementation of the proposed TS changes will allow the affected plant management individuals to use the time now spent in LOR training (i.e., approximately one week out of every six week period throughout the year) to increase their involvement in plant operational matters and planning activities. Therefore, the proposed TS change does not reduce the margin of safety.

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

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

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: Larry E. Nicholson, Acting

Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Dockets Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: October 27, 1993

Description of amendment request: The licensee proposes to amend the Technical Specifications (TS) to allow one of the required on-shift Senior Reactor Operator (SRO) positions to be combined with the required Shift Technical Advisor (STA) position (i.e., dual-role SRO/STA position). The proposed change will permit the licensee to continue to satisfy the NRC policy for engineering expertise on shift, using either of the options discussed in Generic Letter 86-04, "Policy Statement on Engineering Expertise on Shift," dated February 13, 1986.

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 because implementation of the proposed changes will not involve any physical changes to plant SSC [systems, structures or components] or the manner in which these SSC are operated, maintained, modified, tested, or inspected. Therefore, the proposed use of the dual-role SRO/STA position does not increase the probability of an accident previously evaluated.

The consequences of an accident previously evaluated could be affected by the performance of the individual filling the dual-role SRO/STA position. However, implementation of the proposed change will result in personnel with enhanced operational knowledge being assigned to perform the STA function of providing accident assessment expertise and analyzing and responding to off normal occurrences when needed. The NRC's stated preference in the October 28, 1985, "Policy Statement on Engineering Expertise on Shift," indicates that the NRC has concluded that the individual filling the dual-role SRO/STA position may perform these functions better than a non-licensed individual filling the STA position even when the SRO/STA is concurrently functioning as one of the required shift SROs. Furthermore, implementation of the proposed changes will not affect the staffing or qualification of the fire brigade members. Therefore, the proposed TS changes do not increase the consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because implementation of the proposed TS changes will not involve physical changes to plant SSC, or the addition of new SSC. Furthermore, implementation of the proposed changes will not adversely affect the manner in which plant SSC are operated, maintained, modified, tested, or inspected. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. The proposed changes do not involve a significant reduction in a margin of safety because the STA and fire brigade leader positions will be filled by appropriately qualified personnel and shift staffing required by TS Table 6.2.1 and 10CFR50.54(m)(2) is maintained.

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

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

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: Larry E. Nicholson, Acting

Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Dockets Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: November 1, 1993

Description of amendment request: The proposed changes concern the Radiation Monitoring Systems - Isolation and Initiation Functions section of the Technical Specifications (TS) and are necessary to support a plant modification (Mod. 5281). The modification updates the obsolete control room ventilation radiation monitoring equipment and replaces it with a microprocessor based in-duct system.

The proposed administrative change to the Seismic Monitoring Instrumentation section of the TS revises page 240v (Table 4.15), to change the title of Item 3 from "Triaxial response-Spectrum Recorders," to "Central Recording and Analysis System." This will then be consistent with Item 3 of page 240u.

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 Control Room Ventilation Intake Radiation Monitoring System does not serve as an initiator or contributor to any accidents previously evaluated. The system provides indication and detection of radioactivity in the control room ventilation intake and initiates the appropriate trip logic to start the Control Room Emergency Ventilation (CREV) system. This modification increases the number of radiation monitors and reduces the overall complexity of the Control Room Ventilation Intake Radiation Monitoring System. The logic to initiate CREV is revised from one out of two to one out of two twice, to reduce the number of spurious initiations of CREV.

The proposed seismic monitoring changes are purely administrative and will correct an omission from a previously approved TSCR.

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 evaluated.

The proposed Control Room Ventilation Intake Radiation Monitoring System changes support modification 5281 which upgrades the Control Room Ventilation Intake Radiation Monitoring System. The modification replaces the obsolete Control Room Ventilation Intake Radiation Monitoring System equipment with state-of-the-art equipment. All radiation detectors and monitoring components shall have equal or better performance specifications and qualification requirements than the existing components. The new equipment to be installed under modification 5281 does not introduce any new failure modes as compared to the existing equipment.

The proposed seismic monitoring changes are purely administrative and will correct an omission from a previously approved TSCR.

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

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

The current PBAPS Technical Specifications require a minimum of one (1) detector for indication and alarm of radioactive air being drawn into the Control Room be operable. The associated Bases also state that "control room intake air filtration is initiated when a trip signal from the detectors is given." Currently, CREV is initiated via high radiation signals from either detector (using a one out of two logic) or failure signals from both detectors or failure of one detector and low flow in the other detector sample line or low flow in both detector sample lines.

With the new system, CREV will be initiated on 1) high radiation (using a one out of two twice logic), 2) low flow in the control ventilation duct, 3) loss of power in one division at the local radiation monitoring system (RMS) panel, or 4) downscale/failure of the RIS (using a one out of two twice logic). High radiation, low flow in the ventilation duct, loss of power or downscale/failure of an RIS will be annunciated in the control room.

The proposed seismic monitoring changes are purely administrative and will correct an omission from a previously approved TSCR.

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

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

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

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: Larry E. Nicholson, Acting

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

Date of amendment request: October 29, 1993

Description of amendment request: The licensee has requested an amendment to the Technical Specifications (TS) to revise Section 3.10 (Control Rods and Power Distribution Limits) to correct an administrative error that resulted from the issuance of TS

Amendment No. 103. Specifically,

Amendment No. 103, which was issued on September 11, 1990, relocated Figures 3.10-2 (Hot Channel Factor Normalized Operating Envelope) and 3.10-4 (Control Rod Insertion Limits) from the TS to the Core Operating Limit Report (COLR). However, these figures and references to them were not removed from TS. The licensee's amendment request will correct this administrative error and further clarify the TS.

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

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

Response:

The proposed changes do not involve a significant increase in the probability or consequences of any accident previously evaluated. The proposed changes are administrative in nature -- aiming to provide clarity on the status of technical specification figures. The changes do not affect plant system operations, functions, or procedures.

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

Response:

The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated since they are administrative in nature. The changes do not introduce new systems, equipment or procedures.

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

Response:

The proposed changes do not involve significant reductions in margins of safety. The changes are administrative in nature -- clarifying the status of technical specification figures. The changes do not affect system operations, functions, procedures or setpoints.

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

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

Attorney for licensee: Mr. Charles M. Pratt, 10 Columbus Circle, New York, New York 10019.

NRC Project Director: Robert A. Capra

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

Date of amendment request: October 18, 1993

Description of amendment request: The proposed amendment extends the surveillance test intervals (STI's) and allowed out-of-service times (AOTs) for selected instrumentation.

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

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

To justify the STI and AOT relaxation for the selected instrumentation mentioned above, GENE-770-06-1-A (Reference 1) [see October 18, 1993, application] demonstrates the similarity in components, configuration, and function with previously reviewed instrumentation for which STI and AOT relaxations were approved. The analysis for the previously approved STI and AOT relaxation approvals are in NEDC-30851P-A, NEDC-31677P-A (References 3 and 4, respectively) [see October 18, 1993, application]. When all contributing factors are considered, the net impact of the proposed changes is to improve plant safety. These generic analyses have been verified to be applicable to the [Hope Creek Generating Station] HCGS as indicated in Section III above. [See October 18, 1993, application.] Since the proposed changes have a net beneficial impact on plant safety when all factors are considered, the proposed changes will not significantly increase the probability or consequences of a previously analyzed accident.

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

Increasing the AOTs and STIs for the selected instrumentation does not alter the function of the equipment nor involve any type of plant modification. Additionally, no new modes of plant operation are involved with these changes. The proposed changes therefore will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

As requested by the BWR Owners' Group, GE performed analyses to evaluate the effect of the proposed changes on plant safety. The NRC staff has reviewed and approved the generic studies contained in GE LTRs [Licensing Topical Reports] NEDC-30851P-A, NEDC-31677P-A, and GENE-770-06-1-A and has concurred with the BWR Owners Group that the proposed changes do not significantly affect the plant safety. Furthermore, the overall level of plant safety will be improved by the proposed changes. It can therefore be concluded that the proposed changes will not significantly reduce 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: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

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

NRC Project Director: Larry E. Nicholson, Acting

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

Date of amendment request: October 18, 1993

Description of amendment request: The proposed amendment extends the surveillance test intervals (STIs) and allowed out-of-service times (AOTs) for the isolation actuation instrumentation at the Hope Creek Generating Station.

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

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

The proposed changes to the isolation actuation instrumentation were judged to potentially affect plant safety through their impact on the isolation failure frequency (IFF). The generic analyses contained in Licensing Topical Report (LTR) NEDC-30851P-A, Supplement 2 and LTR NEDC-

31677P-A assessed the impact of changing the isolation actuation instrumentation surveillance test intervals (STIs) and allowed out-of-service times (AOTs) on the IFF. The analyses contained in these LTRs demonstrate that the proposed changes have a negligible effect on the IFF, and when all contributing factors are considered, the net impact of the proposed changes is to improve plant safety. These generic analyses have been verified to be applicable to the HCGS [Hope Creek Generating Station] as indicated in Section III above. [See October 18, 1993, application]. Since the proposed changes do not significantly affect the IFF and have a beneficial impact on plant safety when all factors are considered, the proposed changes will not significantly increase the probability or consequences of a previously analyzed accident.

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

Increasing the AOTs and STIs for the isolation actuation instrumentation does not alter the function of the equipment performing the isolation functions nor involve any type of plant modification. Additionally, no new modes of plant operation are involved with these changes. The proposed changes therefore will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes to the isolation actuation instrumentation were judged to potentially affect plant safety through their impact on the IFF. As requested by the BWR Owners' Group, GE performed analyses to evaluate the effect of the proposed changes on the IFF. The NRC staff has reviewed and approved the generic study contained in LTRs NEDC-30851P-A, Supplement 2 and NEDC-31677P-A and has concurred with the BWR Owners Group that the proposed changes do not significantly affect the IFF. Furthermore, the overall level of plant safety will be improved by the proposed changes. It can therefore be concluded that the proposed changes will not significantly reduce 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: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

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

NRC Project Director: Larry E. Nicholson, Acting

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

Date of amendment request: September 30, 1993 (TS 345)

Description of amendment request: The proposed amendment would delete conditions from the Browns Ferry Units 1, 2, and 3 licenses which require maintenance of positive access controls for the containment in accordance with 10 CFR 73.55(d)(8), and deletes a redundant condition from the Unit 3 license.

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

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

The proposed administrative change to the operating licenses does not involve any physical alterations of plant configuration, changes to setpoints, or changes to any operating parameters. The proposed change does not increase the frequency of the precursors to design basis events or operational transients analyzed in the Browns Ferry Final Safety Analysis Report. The change does not alter the designation of BFN [Browns Ferry Nuclear Plant] containment as a vital area, or alter the NRC-approved measures set forth in the BFN Physical Security Plan pertaining to the requirements of 10 CFR 73.55(d)(8). Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed administrative change to the operating licenses does not change any security requirements currently in place at BFN. The proposed change does not alter the requirement to comply with 10 CFR 73.55(d)(8). The change only deletes a duplicative license condition and removes a statement which is no longer necessary to ensure compliance with the requirements of 10 CFR 73.55(d)(8). 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.

The proposed administrative change to the operating licenses does not change or reduce the effectiveness of any security/safeguards measures currently in place at BFN. The proposed change would not remove the requirement to comply with 10 CFR 73.55(d)(8). Therefore, the proposed change does not involve a significant reduction in a 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: Athens Public Library, South Street, Athens, Alabama 35611

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

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: October 27, 1993

Description of amendment request: The proposed amendment adds a footnote to Technical Specification 4.6.1.2.a to allow a one time extension of the test interval for the Type A overall integrated containment leakage rate surveillance. The extension would allow the third Type A test of the first 10-year service period to be delayed until the eighth refueling outage but no later than March 31, 1996. The extension would allow the third test to be performed approximately 54 months after the second test instead of the currently allowed maximum period of 50 months.

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

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

This exemption applies to the ILRT [integrated leak rate testing] and does not affect the local leak rate testing of containment penetrations and isolation valves where the majority of the leakage occurs. The allowable containment leakage used in the accident analysis for offsite doses, L_a, is 0.2 wt %/day and for conservatism the leakage is limited to 75% L_a to account for the possible degradation of containment leakage barriers between tests. Based on the "as-left" leakage data for the past two ILRTs, the additional time period added to the testing interval would not adversely impact the containment leakage barriers to where degradation would cause leakage to exceed that assumed in the accident analysis.

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

There are no design changes being made that would create a new type of accident or malfunction and the method and manner of plant operation remain unchanged. The change to the Surveillance Requirement is a one time exemption to extend the surveillance interval for performance of the third ILRT.

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

There are no changes being made to the safety limits or safety system settings that would adversely impact plant safety. The change is a one time exemption to extend the time interval for performing a ILRT approximately 4 months beyond the current maximum interval. This change does not reduce any technical specification 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 Locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621

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

NRC Project Director: Suzanne C. Black

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.

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

Date of amendment request: November 4, 1993, as supplemented November 4, 1993.

Description of amendment request: The proposed amendment would increase the required supplementary leak collection and release system (SLCRS) drawdown time from 60 seconds to 120 seconds and increase the required vacuum to 0.4 inches, based on compensating reductions in containment leak rate. Date of publication of individual notice in **Federal Register**: November 12, 1993 (58 FR 60072)

Expiration date of individual notice: December 13, 1993

Local Public Document Room location: Learning Resources Center, Thames Valley State Technical College, 574 New London Turnpike, Norwich, Connecticut 06360.

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 20555, and at the local public document rooms for the particular facilities involved.

Arizona Public Service Company, et al., Docket No. 50-528, Palo Verde Nuclear Generating Station, Unit 1, Maricopa County, Arizona

Date of application for amendment: September 8, 1993

Brief description of amendment: The amendment adds a methodology supplement entitled, "System 80™ Inlet Flow Distribution," to the list of methods used to determine the core operating limits.

Date of issuance: November 19, 1993

Effective date: November 19, 1993

Amendment No.: 72

Facility Operating License No. NPF-41: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 15, 1993 (58 FR 53585)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 19, 1993.

No significant hazards consideration comments received: No.

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

Commonwealth Edison Company, Docket Nos. 50-295 and 50-304, Zion Nuclear Power Station Units 1 and 2, Lake County, Illinois

Date of application for amendments: April 27, 1993

Brief description of amendments: The amendments revise the reactor protection and engineered safeguards and limiting safety system settings of the Technical Specifications by: (1) adding steam generator overfill protection requirements, and (2) modifying the equations for the overpower delta T (OPDT) and overtemperature delta T (OTDT) protective functions.

Date of issuance: November 15, 1993

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

Amendment Nos.: 150 and 138

Facility Operating License Nos. DPR-39 and DPR-48. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 13, 1993 (58 FR 52981)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 15, 1993.

No significant hazards consideration comments received: No

Local Public Document Room location: Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085.

Connecticut Yankee Atomic Power Company, Docket No. 50-213, Haddam Neck Plant, Middlesex County, Connecticut; and Northeast Nuclear Energy Company, Docket Nos. 50-245, 50-336, and 50-423, Millstone Nuclear Power Station, Units 1, 2, and 3, New London County, Connecticut

Date of application for amendments: July 16, 1993

Brief description of amendments: The amendments revise the Technical Specifications to change the submittal frequency of the Radioactive Effluent Release Report from semiannual to annual to be submitted by May 1 of each year, and also, consolidates the Radioactive Effluent Release Report and the Radioactive Effluents Dose Report into a single annual report entitled Radioactive Effluent Report.

Date of issuance: November 23, 1993

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

Amendment Nos.: 170, 69, 169, and 86

Facility Operating License Nos. DPR-61, DPR-21, DPR-65, and NPF-49. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** September 1, 1993 (58 FR 46226)

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated November 23, 1993.

No significant hazards consideration comments received: No.

Local Public Document Room location: Russell Library, 123 Broad Street, Middletown, Connecticut 06457 for the Haddam Neck Plant; and the Learning Resources Center, Thames Valley State Technical College, 574 New London Turnpike, Norwich, Connecticut 06360 for Millstone Units 1, 2, and 3.

Consumers Power Company, Docket No. 50-155, Big Rock Point Plant, Charlevoix County, Michigan

Date of application for amendment: August 6, 1993

Brief description of amendment: The amendment changes the Technical Specifications to implement a reorganization of the Big Rock Point staff.

Date of issuance: November 15, 1993

Effective date: November 15, 1993

Amendment No.: 112

Facility Operating License No. DPR-6. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** October 13, 1993 (58 FR 52983).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 15, 1993.

No significant hazards consideration comments received: No.

Local Public Document Room location: North Central Michigan College, 1515 Howard Street, Petoskey, Michigan 49770.

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

Date of amendment request: August 5, 1993

Brief description of amendment: The amendment revised the Technical Specifications for the Containment Spray System to clarify the requirements for Applicability in Mode 4 and to increase the testing interval for verifying that each containment spray nozzle is unobstructed.

Date of issuance: November 17, 1993

Effective date: November 17, 1993

Amendment No.: 89

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

Date of initial notice in **Federal Register:** September 15, 1993 (58 FR 48383)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 17, 1993.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: September 7, 1993, as supplemented September 24, 1993

Brief description of amendment: The amendment revised Technical Specifications for the incore detection system to allow less than 75% but more than 50% of the incore locations to be operable provided the appropriate penalties are applied to the core operating limit supervisory system (COLSS) and the core protection calculators (CPCs). This change is effective for the remainder of the current Fuel Cycle 6.

Date of issuance: November 18, 1993

Effective date: November 18, 1993

Amendment No.: 90

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

Date of initial notice in **Federal Register:** October 13, 1993 (58 FR 52984)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 18, 1993.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: October 21, 1992

Brief description of amendment: The amendment revised the Technical Specifications on component cooling water (CCW) radiation monitors to clearly distinguish between the monitors and to remove the requirement for monitor A/B during Modes 5 and 6 where operation is difficult due to low flow in the CCW line from containment.

Date of issuance: November 22, 1993

Effective date: November 22, 1993

Amendment No.: 91

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

Date of initial notice in **Federal Register:** November 25, 1992 (57 FR 55580)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 22, 1993.I11No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 20, 1993

Brief description of amendments: These amendments implement new 10 CFR Part 20 requirements relating to radiological effluent releases, and change the frequency of reporting the release of radioactive effluents from semi-annual to annual.

Date of issuance: November 18, 1993

Effective date: November 18, 1993Amendment Nos. 157 and 151Facility Operating Licenses Nos. DPR-31 and DPR-41: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 18, 1993 (58 FR 43926)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 18, 1993.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 7, 1993, as supplemented on October 1, 1993.

Brief description of amendment: This amendment modifies Technical Specification (TS) 4.6.A, Safety Injection and Containment Spray Systems, to: 1) require quarterly, vice monthly, testing of automatic core flooding and containment spray valves, 2) require that containment isolation valves not tested quarterly during reactor operation be tested during the next refueling outage, and 3) require an air flow test of all containment spray nozzles every 10 years, twice every 5 years. This amendment also modifies TS 4.6.B, Emergency Feedwater Pumps, to ...re quarterly, vice monthly testing ...ergency and auxiliary feedwater ...ps. Finally, minor editorial changes are made in TS 4.6.A and B to clarify existing requirements.

Date of issuance: November 5, 1993

Effective date: November 5, 1993

Amendment No.: 143

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

Date of initial notice in **Federal Register**: July 21, 1993 (58 FR 39053)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 5, 1993.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: May 7, 1993, as superseded September 28, 1993.

Brief description of amendment: The amendment adds a new Technical Specification (TS) 3/4.10.7, "Inservice Leak and Hydrostatic Testing," to Nine Mile Point Nuclear Station, Unit 2, TSs. The amendment also includes corresponding changes to the TS Index, Table 1.2, and provides Bases for TS 3/4.10.7. The added TS 3/4.10.7 permits the unit to remain in OPERATIONAL CONDITION 4 with average reactor coolant temperature being increased above 200°F during reactor coolant system inservice leak or hydrostatic tests provided the maximum reactor coolant temperature does not exceed 212°F and the following OPERATIONAL CONDITION 3 TSs are being met: (a) TS 3.3.2, "Isolation Actuation Instrumentation," Functions 1.a.2, 1.b, and 3.a and b of Table 3.3.2-1; (b) TS 3.6.5.1, "Secondary Containment Integrity;" (c) TS 3.6.5.2, "Secondary Containment Automatic Isolation Dampers;" and (d) TS 3.6.5.3, "Standby Gas Treatment System."

Date of issuance: November 12, 1993

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

Amendment No.: 53

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

Date of initial notice in **Federal Register**: June 9, 1993 (58 FR 32386) and renoticed October 13, 1993 (58 FR 52990)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 12, 1993.

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.

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

Date of application for amendments: July 16, 1993

Brief description of amendments: The amendments revise the Technical Specifications contained in Appendix A of the Operating Licenses, to allow one of the required on-shift Senior Reactor Operator positions to be combined with the required Shift Technical Advisor position.

Date of issuance: November 15, 1993

Effective date: November 15, 1993Amendment Nos. 64 and 29

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

Date of initial notice in **Federal Register**: September 15, 1993 (58 FR 48387)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 15, 1993.

No significant hazards consideration comments received: No

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

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

Date of application for amendments: August 20, 1993

Brief description of amendments: These amendments revised the surveillance requirements for the standby gas treatment system (SGTS) charcoal filter deluge system. The revised surveillance requirements reflect a planned modification of the deluge system actuation from an automatic to a manual operation.

Date of issuance: November 16, 1993

Effective date: As of its date of issuance and shall be implemented within 90 days of the date of issuance.Amendments Nos.: 181 and 186

Facility Operating License Nos. DPR-44 and DPR-56: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 15, 1993 (58 FR 48387)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 16, 1993.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: November 19, 1992, and supplemented December 29, 1992, May 28, 1993, and September 3, 1993

Brief description of amendment: The amendment revises Public Service Electric and Gas Company's (PSE&G) commitments in two Updated Final Safety Analysis Report (UFSAR) sections. Specifically, the amendment relieves PSE&G from its commitment to fully comply with the Emergency Diesel Generator fuel oil storage recommendations in Standard Review Plan Section 9.5.4, Paragraph I.1.d and Regulatory Guide 1.137, Revision 1.

Date of issuance: November 22, 1993

Effective date: November 22, 1993

Amendment No.: 59

Facility Operating License No. NPF-57: This amendment revised the UFSAR.

Date of initial notice in **Federal Register:** February 17, 1993 (58 FR 8779)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 22, 1993.

No significant hazards consideration comments received: No

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

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

Date of application for amendments: July 19, 1993, and supplemented by letter dated August 5, 1993

Brief description of amendments: The amendments delete Line Item 9, Boric Acid Tank Solution Level, from Tables 3.3-11 and 4.3-11 and the associated Action 3 of Technical Specification 3.3.3.7, Post Accident Monitoring System.

Date of issuance: November 16, 1993

Effective date: November 16, 1993

Amendment Nos. 147 and 125

Facility Operating License Nos. DPR-70 and DPR-75. These amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** September 1, 1993 (58 FR 42640)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 16, 1993.

No significant hazards consideration comments received: No

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, New Jersey 08079

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: March 10, 1993 (TS 92-08)

Brief description of amendments: The amendments incorporate the technical specification changes necessary to reduce the boric acid concentration in the boric acid tanks to be reduced from 12 percent to approximately 3.5 to 4.0 percent.

Date of issuance: November 26, 1993

Effective date: November 26, 1993

Amendment Nos.: 172 - Unit 1; 163 - Unit 2

Facility Operating License Nos. DPR-77 and DPR-79: Amendments revise the Technical Specifications.

Date of initial notice in **Federal Register:** May 12, 1993 (58 FR 28058)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 26, 1993.

No significant hazards consideration comments received: No

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

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

Date of amendment request: May 28, 1993, as supplemented by letter dated September 24, 1993.

Brief description of amendment: The amendments change the technical specifications by incorporating changes for Cycle 4 operations in Unit 1; specifically, to allow the use of additional NRC-approved methodologies and to revise core safety limit curves and N-16 overtemperature reactor trip setpoints. In addition, the amendments increase the minimum required reactor coolant system flow, remove a penalty on pressurizer pressure uncertainty, and include an operational enhancement for the treatment of the uncertainty allowance for the N-16 power indication.

Date of issuance: November 16, 1993

Effective date: November 16, 1993, to be implemented within 30 days of issuance.

Amendment Nos: Unit 1 - Amendment No. 21; Unit 2 - Amendment No. 7

Facility Operating License Nos. NPF-87 and NPF-89: Amendments revised the Technical Specifications

Date of initial notice in **Federal Register:** August 18, 1993 (58 FR 43934). The September 24, 1993, submittal provided supplemental information to the application and did not change the initial no significant hazards determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 16, 1993.

No significant hazards consideration comments received: No

Local Public Document Room location: University of Texas at Arlington Library, Government Publications/Maps, 701 South Cooper, P. O. Box 19497, Arlington, Texas 76019

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

Date of application for amendment: June 23, 1993, as supplemented on October 5, 1993

Brief description of amendment: The amendment allows storage of new and spent fuel assemblies with an initial enrichment of Uranium-235 no greater than 5.0 weight percent.

Date of issuance: November 19, 1993

Effective date: November 19, 1993

Amendment No. 181

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

Date of initial notice in **Federal Register:** August 4, 1993 (58 FR 41516) The supplemental letter provided additional information that did not change the initial proposed no significant hazard consideration determination.

The Commission's related evaluation of the amendment is contained in an Environmental Assessment dated November 1, 1993, and in a Safety Evaluation dated November 19, 1993.

No significant hazards consideration comments received: No

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

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 20555, 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 January 7, 1994, 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 20555 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, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (**Project Director**): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

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

Omaha Public Power District, Docket No. 50-285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: June 17, 1993, as supplemented October 8, 1993.

Brief description of amendment: The amendment implemented administrative changes. The changes include providing consistency with Combustion Engineering Standard Technical Specifications on refueling frequency, incorporating bases information on pressurizer safety valves, correcting typographical and grammatical problems, and correcting mistakes in previous amendments.

Date of issuance: November 22, 1993

Effective date: November 22, 1993

Amendment No.: 157

Facility Operating License No. DPR-40. Amendment revised the Technical Specifications.

Public comments requested to proposed no significant hazards consideration: Yes, August 4, 1993 (58 FR 41509) and November 8, 1993 (58 FR 59280).

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 November 22, 1993.

Attorney for licensee: LeBoeuf, Lamb, Leiby, and MacRae, 1875 Connecticut Avenue, N.W., Washington, D.C. 20009-5728

Local Public Document Room location: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102

NRC Project Director: William D. Beckner

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

Date of application for amendments: November 10, 1993, as supplemented November 16, 1993.

Brief description of amendments: The amendments eliminate the simulated reactor coolant pump seal injection flow requirement for the flow balancing of the high head safety injection lines.

Date of issuance: November 23, 1993

Effective date: November 23, 1993

Amendment Nos.: 176 and 157

Facility Operating License Nos. NPF-4 and NPF-7: Amendments revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: No. Verbally, on November 8, 1993, and by letter dated November 10, 1993, the staff granted an enforcement discretion to be in effect until the amendments were issued.

The Commission's related evaluation of the amendments, consultation with the State of Virginia and final no significant hazards determination are contained in a safety evaluation dated November 23, 1993.

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

Local Public Document Room location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903-2498.

Dated at Rockville, Maryland, this 1st day of December 1993.

For the Nuclear Regulatory Commission

Steven A. Varga,

Director, Division of Reactor Projects - I/II, Office of Nuclear Reactor Regulation.

[Doc. 93-29808 Filed 12-7-93; 8:45]

BILLING CODE 7590-01-F

Printed
on recycled
paper

Federal Recycling Program

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

FIRST CLASS MAIL
POSTAGE AND FEES PAID
USNRC
PERMIT NO. G 67