[Federal Register: November 9, 1994]

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from October 17, 1994, through October 28, 1994. The last biweekly notice was published on October 26, 1994 (59 FR 53834).

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

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

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

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

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

By December 9, 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.

Carolina Power & Light Company, et al.

Docket Nos. 50-325 and 50-324

Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina.

Date of amendments request: September 30, 1994.

Description of amendments request: The amendments would revise the Technical Specifications to eliminate the scram and isolation trip functions from the main steam line radiation monitor (MSLRM). This change would specifically remove the reactor scram, main steam line isolation valve closure, main steam line drain valve closure, reactor water sample line isolation, and mechanical vacuum pump line isolation actuated on a MSLRM High-High Radiation signal. The actuation signal for isolation of the reactor water sample line will be replaced with a low condenser vacuum signal. The isolation of the mechanical vacuum pump line will be changed to a signal from the main stack radiation monitor.

The MSLRMs will have both High Radiation and High-High Radiation alarms. The setpoint for the MSLRM High Radiation alarm will be set at or below 1.5 times the nominal full power background radiation adjusted
for Hydrogen water chemistry operation. The setpoint for the condenser

off-gas radiation monitor will be set at a value of 1.5 times background radiation, but not less than 1.5 Rem per hour.

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 amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated. The deletion of the MSLRM trip function from the reactor scram and the Group 1 isolation initiation logic removes a potential transient initiation and therefore decreases the probability of plant transients occurring due to inadvertent scrams resulting from this system.

The deletion of the MSLRM trip function from the Main Steam Drain Valve, the Reactor Water Sample Isolation Valve, and the Mechanical Vacuum Pump line isolation logic, does not affect the initiators of any accident previously evaluated in the Safety Analysis Report. Therefore, the proposed change does not involve an increase in the probability of occurrence of any accident previously evaluated.

The NRC staff acceptance criterion for the Control Rod Drop Accident is that the doses from the accident fall significantly below the limits given in 10 CFR Part 100. The releases calculated for accident during plant operations when the Steam Jet Air Ejectors (SJAE) are operating and when the Mechanical Vacuum Pumps are operating are within these acceptance limits.

In NEDO-31400, GE shows that the occurrence of a CRDA, with the MSL high radiation isolation removed, and SJAE in operation, results in offsite radiological exposures that are small fractions of 10CFR100 guidelines. Since the Brunswick specific CRDA doses are lower than the [sic] calculated by GE and the GE dose parameters envelope those used for the Brunswick analysis, it is concluded that the NRC's findings that the radiological release consequence is within the staff's acceptance criteria, even without the automatic MSIV trip, is applicable to Brunswick.

While not specifically addressed in the GE evaluation, Carolina Power and Light also proposes to eliminate the Main Steam Line Drain valves, the Reactor Water Sample Line isolation valves, and the mechanical vacuum line isolation valves from the MSLRM isolation logic.

Main Steam Line Drain Valves B21-F016 and B21-F019 drain to the main condenser, which is the same flow path as the MSIVs. The discharge of both the MSIV and MSL drain flow paths is processed through the offgas system. Any radiation released through the drain valves during a control rod drop accident will be negligible and, for Brunswick, is bounded by the NEDO analysis.

The reactor water sample line provides a small amount of reactor water to the Reactor Building Sample Panel. The discharge of the Reactor Building Sample Panel is routed through the floor drain sump to the liquid radwaste system. Any releases through this path would be negligible and, for Brunswick, is bounded by the NEDO analysis.

The mechanical vacuum pumps are used only when the reactor is at low power (less than 5%) and there is insufficient steam flow to operate the Steam Jet Air Ejectors. The increase in radiation will be detected by the MSLRMs and annunciated in the Main Control Room. Operators will be instructed, in the annunciator response procedures, to take action to stop the Mechanical Vacuum Pump(s) and isolate the Mechanical Vacuum Pump line. The amount of radiation released prior to isolating the line would represent the most limiting case for this accident. However, it will still be well within 10 CFR Part 100 limits.
Additionally, the dose received in the Main Control Room as a result of this accident is within General Design Criteria 19 (SRP 6.4) limits.

Therefore, since elimination of the MSIV [sic, MSLRM] scram and isolation functions would not result in an increase in exposure above NRC acceptance limits, the proposed changes will not significantly increase the consequences of a previously evaluated accident.

2. The proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated. The function of a MSLRM trip is to detect abnormal fission product release and isolate the steam lines, thereby stopping the transport of fission products from the reactor to the main condenser. The monitors do not perform a prevention function for any kind of accident. The existence of a MSLRM trip does not prevent the occurrence of a fuel failure event or any other type of event. The elimination of these signals, which served only in a mitigative function, does not create the possibility of a new or different kind of accident from those previously evaluated. Also, radiation monitors with alarm functions will remain installed in the plant to warn the operators of a

high radiation condition in the main steam lines, or in the off-gas system. Thus **no** new or different accident can be postulated by the proposed changes.

3. The proposed amendments do not involve a significant reduction in a margin of safety. As shown in the topical report, the changes represent an overall improvement in plant safety. Safe operation of the plant is further enhanced by elimination of the unnecessary scram and isolation of the reactor vessel. With implementation of these changes, 1) the primary heat sink (main condenser) remains available, 2) large transients on the reactor vessel, as well as challenges to the ESF, are avoided, and 3) the Offgas system remains available to control the pathway of potential releases. As such, the margin of safety is enhanced by the proposed changes.

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

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

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

NRC Acting Project Director: Michael L. Boyle.

Carolina Power & Light Company

Docket **No**. 50-261

H. B. Robinson Steam Electric Plant, Unit **No**. 2, Darlington County, South Carolina.

Date of amendment request: October 7, 1994.

Description of amendment request: The proposed amendment would revise the introduction to TS Section 6.9.3.3 to require the approved revision number for the referenced analytical methods be listed in the Core Operating Limits Report. The methodology referenced in 6.9.3.3.b.f (XN-NF-82-49(A)) will be updated to clarify that all supplements are included. New methodologies ANF-89-151(A) and EMF-92-081(A) will be added to TS Section 6.9.3.3.b.

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

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

The proposed changes will have **no** influence on the probability of an accident previously evaluated. **No** changes will be made to any safety related equipment, systems, or setpoints used in determining the probability of an evaluated accident. The plant design basis will not be altered. Therefore, there will be **no** significant increase in the probability of an accident previously evaluated.

Consequences are dependent on the type of accident and the mitigating response of safety related equipment. Furthermore, the magnitude of consequences are calculated, directly or through supporting calculations, by use of NRC approved methodologies. The proposed license amendment will not alter the function of safety related equipment designed to mitigate the consequences of an accident previously evaluated or allow operation of the facility outside any current limitations or restrictions. Also, this amendment will not alter the requirement that evaluation of the consequences of an accident previously evaluated by determined/supported with NRC reviewed and approved methodologies. The change to TS Section 6.9.3.3.b's introductory wording satisfies an administrative commitment and the requirements it adds are administrative in nature. Accordingly the proposed license amendment will not involve a significant increase in 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 addition of and update to NRC previously reviewed and approved methodologies in TS Section 6.9.3.3.b will not result in any design or function changes to any safety related equipment designed to prevent and/or mitigate accidents, to any setpoints or systems, or to any portion of the plant design basis. Operation of the facility will remain within all required limitations and/or restrictions. The change to TS Section 6.9.3.3.b's introductory wording satisfies an administrative commitment and the requirements it adds are administrative in nature. Therefore, the proposed amendment will not create the possibility of a new kind of accident from any accident previously evaluated.

The addition of and update to NRC previously reviewed and approved methodologies in TS Section 6.9.3.3.b will not result in any design or function changes to any safety related equipment designed to prevent and or mitigate accidents, to any setpoints or systems, or to any portion of the plant design basis. Operation of the facility will remain within all required limitations and/or restrictions. The changes to TS Section 6.9.3.3.b's introductory wording satisfies an administrative commitment and the requirements it adds are administrative in nature. Therefore, the proposed amendment will not create the possibility of a 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 license amendment is defined as administrative in nature. **No** current operational limits, restrictions, or operating modes of the facility and its equipment, safety related or otherwise, designed to preserve the margin of safety will be changed or affected by the proposed amendment. There will be **no** changes to setpoints or to the plant design basis. The methodology proposed for addition to TS Section 6.9.3.3.b and the methodology that will be updated has been previously reviewed and approved by the NRC. The change to TS Section 6.9.3.3.b's introductory wording satisfies an administrative commitment and the requirements it adds are administrative in nature. Accordingly the proposed license amendment will not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: William H. Bateman.

Entergy Operations, Inc., et al.

Docket **No**. 50-416

Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi.

Date of amendment request: October 12, 1994.

Description of amendment request: The proposed amendment requests the closure and deletion of License Condition 2.C.(26) related to turbine disk integrity.

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

1. **No** significant increase in the probability or consequences of an
accident previously evaluated results from this change.

The proposed change would close and delete License Condition 2.C.(26). The approved methodology currently used to evaluate the probability of rotor failure and the inspection interval will not be changed. The closure and deletion of the license condition is an administrative change and will affect any accident previously evaluated.

The bounding accident for the turbine-generator as analyzed in the Grand Gulf Nuclear Station (GGNS) Updated Final Safety Analysis Report (UFSAR) is the occurrence of an external missile resulting from the failure of a low pressure (LP) turbine disc. The probability of this incident occurring is less than 1 x 10<SUP>-5 per year, which is the NRC acceptable failure criterion for probability.

Any extension to the service interval in the future will be evaluated in accordance with the current methodology. The original acceptable levels of failure will be maintained. Therefore, **no** significant increase in the probability or consequences of a previously
evaluated accident results from this change.

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

The proposed change does not involve a change to the control logic or operating procedures for the turbine but rather transfers the control of the LP turbine disc inspection interval from the Operating License to administrative control. The current approved methodology will continue to be used when determining future inspection intervals.

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

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

Closing and deleting the current license condition for LP turbine disc inspections and controlling the inspection interval

administratively has **no** adverse effects to the margin of safety. The current approved methodology for failures will continue to be used and any changes to future inspection intervals will be evaluated by the methodology. This change does not affect any previous safety analysis presented in the UFSAR and does not affect the criteria used to establish safety limits, the basis for limiting safety system settings,
the basis for limiting conditions of operation, a change to the technical specifications or a change in plant operations.

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

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

Local Public Document Room Location: Judge George W. Armstrong Library, 220 S. Commerce Street, 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.

Florida Power and Light Company

Docket Nos. 50-250 and 50-251

Turkey Point Plant, Units 3 and 4, Dade County, Florida.

Date of amendment request: October 20, 1994.

Description of amendment request: The licensee proposes to change Turkey Point, Units 3 and 4 Technical Specifications (TS) by revising TS 1.9, Definitions--CORE ALTERATIONS to only address activities which may, in actuality, affect core reactivity. In addition, the licensee proposes to revise TS 3.9.4, Containment Building Penetrations to allow
both containment personnel airlock (PAL) doors to be open during core alterations and movement of irradiated fuel in containment provided (a)
that at least one PAL door is capable of being closed; (b) the plant is
in Mode 6 with at least 23 feet of water above the fuel; and (c) a designated individual is available outside the PAL to close the door. The licensee also proposes a revision to the footnote of TS 3.9.4, to remove the description of the purpose for imposing administrative

controls.

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

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The change in the definition of CORE ALTERATIONS would allow the movement of a temporary source range detector or other small components, such as cameras, tools, etc., within the reactor vessel without the activity being considered CORE ALTERATIONS. The potential exists, however small, that an object can be dropped into the reactor vessel. However, the justification for this change, is that the insertion of small components into the reactor vessel will have **no** effect on core reactivity since these items displace a small volume of borated water, and sufficient borated water will surround the components and provide the necessary neutron absorption to neutronically isolate the components from the reactor. The consequences of dropping one of these small components into the vessel are bounded by the In-Containment Fuel Handling Accident Analysis discussed in Chapter 14.2.1 of the Turkey Point Updated Final Safety Analysis Report (UFSAR). Therefore, the proposed change is bounded by the current and the proposed In-Containment Fuel Handling Accident Analyses and will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to TS 3.9.4 would allow the containment personnel airlock (PAL) doors to be open during fuel movement and core alterations. Currently, a single PAL door is closed during fuel movement and core alterations to prevent the escape of radioactive material in the event of a in-containment fuel handling accident. The PAL is not an initiator of an 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 not increase the consequences from a fuel handling accident. The calculated offsite doses are well within the limits of 10 CFR Part 100. 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
should significantly reduce the dose to workers in containment in the event of a fuel handling accident by reducing the time required to evacuate the containment. 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 to the footnote of TS 3.9.4 is administrative in nature, and does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The changes being proposed do not affect assumptions contained in plant safety analyses or the physical design of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant increase in the
probability or consequences of an accident previously analyzed.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The change in the definition of CORE ALTERATIONS would allow the movement of a temporary source range detector or other small components, such as cameras, tools, etc., within the reactor vessel without the activity being considered CORE ALTERATIONS. The potential exists however small, that an object can be dropped into the reactor vessel. However, the justification for this change, is that the insertion of small components into the reactor vessel will have **no** effect on core reactivity since these items displace a small volume of borated water, and sufficient borated water will surround the components and provide the necessary neutron absorption to neutronically isolate the components from the reactor. The consequences
of dropping one of these small components into the vessel are bounded by the In-Containment Fuel Handling Accident Analysis discussed in Chapter 14.2.1 of the Turkey Point UFSAR. Therefore the proposed change
is bounded by the current and the proposed In-Containment Fuel Handling
Accident Analyses and will not create the possibility of a new or different kind of accident.

The proposed change to Specification 3.9.4 affects a previously evaluated accident, i.e., in-containment fuel handling accident. Both the current and the proposed In-Containment Fuel Handling Accident

Analysis assume that all of the iodines and noble gases that become airborne within the containment escape and reach the site boundary and low population zone with **no** credit taken for the containment building barrier or for decay or deposition taken. Since the proposed change does not involve the addition or modification of equipment nor does it alter the design of plant systems and the revised analysis is consistent with the current In-Containment Fuel Handling Accident Analysis, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change to the footnote of TS 3.9.4 is administrative in nature and does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The change in the definition of CORE ALTERATIONS would allow the movement of a temporary source range detector or other small components, such as cameras, tools, etc., within the reactor vessel without the activity being considered CORE ALTERATIONS. The potential exists however small, that an object can be dropped into the reactor vessel. However, the justification for this change, is that the insertion of small components into the reactor vessel will have **no** effect on core reactivity since these items displace a small volume of borated water, and sufficient borated water will surround the components and provide the necessary neutron absorption to neutronically isolate the components from the reactor. The consequences of dropping one of these small components into the vessel are bounded by the Fuel Handling Accident Analysis discussed in Chapter 14.2.1 of the Turkey Point UFSAR. Therefore, the proposed change is bound by the current In-Containment Fuel Handling Accident Analyses and as a result will not involve a significant reduction in a margin of safety.

The margin of safety as defined by 10 CFR Part 100 has not been reduced. There is **no** increase in calculated offsite dose resulting from a fuel handling accident in containment and the calculated dose is a small fraction of the limits given in 10 CFR Part 100. The proposed changes do not alter the bases for assurance that safety-related activities are performed correctly or the basis for any Technical Specification that is related to the establishment of or maintenance of a safety margin. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant reduction

in a margin of safety.
The proposed change to the footnote of TS 3.9.4 is administrative in nature and does not relate to or modify the safety margins defined in, and maintained by, the Technical Specifications.
The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Florida International University, University Park, Miami, Florida 33199.
Attorney for licensee: Harold F. Reis, Esquire, Newman and Holtzer, P.C., 1615 L Street, NW., Washington, DC 20036.
NRC Project Director: Mohan C. Thadani, (Acting)

Florida Power and Light Company

Docket Nos. 50-250 and 50-251

Turkey Point Plant, Units 3 and 4, Dade County, Florida.
Date of amendment request: October 20, 1994.
Description of amendment request: This supersedes the licensee's original request dated July 19, 1994, and noticed in the Federal Register on August 3, 1994 (59 FR 39588). The licensee proposes to change Turkey Point, Units 3 and 4 Technical Specifications (TS) 4.8.1.1.2e. and 4.8.1.1.2f., which address Emergency Diesel Generator (EDG) fuel oil testing, by replacing the specific EDG fuel oil Surveillance Requirements with the requirement to verify new and stored EDG fuel oil in accordance with the Diesel Fuel Oil Testing Program. In addition, the licensee proposes the addition of ACTION statements g. and h., to TS 3.8.1.1, to address the required action in the event the diesel fuel oil does not meet the Diesel Fuel Oil Testing Program limits. The Diesel Fuel Oil Testing Program will be described in both TS 6.8.4 and the BASES Section to the Technical Specifications. In addition, FPL proposes revising TS 6.8.1 to include the requirement that written procedures shall be established, implemented and maintained for implementation of the Diesel Fuel Oil Testing Program.
Basis for proposed **no** significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of **no** significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to the Technical Specifications will permit the Technical Specification required testing of Emergency Diesel Generator (EDG) fuel oil in accordance with the Turkey Point, Units 3 and 4 Diesel Fuel Oil Testing Program. The proposed change will permit FPL to use more recent editions of the American Society for Testing and Materials (ASTM) standards currently listed in Technical Specification Surveillance Requirements 4.8.1.1.2e. and 4.8.1.1.2f. Prior to changing the Diesel Fuel Oil Testing Program, the proposed change will be evaluated pursuant to Title 10 Code of Federal Regulations Sec. 50.59 (10 CFR Sec. 50.59), ``Changes, tests, and experiments.'' Title 10 CFR Sec. 50.59 permits a licensee to make changes in the procedures as described in the safety analysis report without prior Commission approval, provided that the proposed changes does not involve an unreviewed safety question.

Title 10 CFR Sec. 50.59(a)(2) states that a proposed change involves an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased. Consequently, since any change to the Diesel Fuel Oil Testing Program, including the ASTM standard or ASTM edition standard to be used to evaluate EDG fuel oil acceptability, the change must be evaluated relative to the more restrictive evaluation criterion of 10 CFR Sec. 50.59, then operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated. The EDG fuel oil TS Surveillance Requirements will be replaced with a requirement to test the EDG fuel oil in accordance with the Turkey Point Units 3 and 4 Diesel Fuel Oil Testing Program.

ACTION statement g. of TS 3.8.1.1 is added to address the required action in the event the new fuel oil properties do not meet the Diesel Fuel Oil Testing Program limits. A failure to meet the American Petroleum Institute (API) gravity, kinematic viscosity, flash point or clarity limits is cause for rejecting the new fuel oil prior to the addition to the Diesel Fuel Oil Storage Tanks, but does not represent

a
failure to meet the Limiting Condition for Operation (LCO) of TS 3.8.1.1, since the new fuel oil has not been added to the storage tanks. Provided these new fuel oil properties are met subsequent to the
addition of the new fuel oil to the storage tanks, 30 days is provided to complete the analyses of the other fuel oil properties specified in Table 1 of ASTM-D975-81, except sulfur which may be performed in accordance with ASTM-D1552-79 or ASTM-D2622-82. In the event the other new fuel oil properties specified in Table 1 of ASTM-D975-81 are not met, ACTION statement g. of TS 3.8.1.1 provides an additional 30 days to meet the Diesel Fuel Oil Testing Program limits. This additional 30 day period is acceptable because the fuel oil properties of interest, even if they are not within limits, would not have an immediate effect on EDG operation.

ACTION statement h. of TS 3.8.1.1 is added to address the required action in the event the stored fuel oil total particulates do not meet the Diesel Fuel Oil Testing Program limits. Fuel oil degradation during
long term storage shows up as an increase in particulate, due mostly to
oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The frequency for performing surveillance on stored fuel oil is based on stored fuel oil degradation
trends which indicate that particulate concentration is unlikely to change significantly between surveillances.

Prior to changing the Turkey Point Units 3 and 4 Diesel Fuel Oil Testing Program, FPL will need to determine if the proposed program change is at least as, if not more, effective, in detecting unsatisfactory fuel oil. The EDGs will thus continue to function as designed and the probability or consequences of previously evaluated accidents will be unaffected.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to the Technical Specifications will permit the Technical Specification required testing of Emergency Diesel Generator fuel oil using more recent editions of the American Society for Testing and Materials (ASTM) standards currently listed in Technical Specification Surveillance Requirements 4.8.1.1.2e. and 4.8.1.1.2f. Prior to changing the edition of the previously approved ASTM standard being used to evaluate the EDG fuel oil, the proposed

edition standard will be evaluated pursuant to 10 CFR Sec. 50.59, ``Changes, tests, and experiments.'' Title 10 CFR Sec. 50.59 permits a licensee to make changes in the procedures as described in the safety analysis report without prior Commission approval, provided that the proposed changes does not involve an unreviewed safety question. Title 10 CFR Sec. 50.59(a)(2) states that a proposed change involves an unreviewed safety question (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created. Consequently, since any change to the edition of the ASTM standard to be used to evaluate EDG fuel oil
acceptability must be evaluated relative to the more restrictive evaluation criterion of 10 CFR Sec. 50.59, then operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

ACTION statement g. of TS 3.8.1.1 is added to address the required action in the event the new fuel oil properties do not meet the Diesel Fuel Oil Testing Program limits. A failure to meet the API gravity, kinematic viscosity, flash point or clarity limits is cause for rejecting the new fuel oil prior to the addition to the Diesel Fuel Oil
Storage Tanks, but does not represent a failure to meet the Limiting Condition for Operation (LCO) of TS 3.8.1.1, since the new fuel oil has
not been added to the storage tanks. Provided these new fuel oil properties are met subsequent to the addition of the new fuel oil to the storage tanks, 30 days is provided to complete the analyses of the other fuel oil properties specified in Table 1 of ASTM-D975-81, except sulfur which may be performed in accordance with ASTM-D1552-79 or ASTM-D2622-82. In the event the other new fuel oil properties specified in Table 1 of ASTM-D975-81 are not met, ACTION statement g. of TS 3.8.1.1 provides an additional 30 days to meet the Diesel Fuel Oil Testing Program limits. This additional 30 day period is acceptable because the
fuel oil properties of interest, even if they are not within limits, would not have an immediate effect on EDG operation.

ACTION statement h. of TS 3.8.1.1 is added to address the required action in the event the stored fuel oil total particulates does not meet the Diesel Fuel Oil Testing Program limits. Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The frequency for performing surveillance on stored fuel oil is based on stored fuel oil degradation trends which indicate that particulate concentration is unlikely to change significantly between surveillances.

Prior to changing the Turkey Point Units 3 and 4 Diesel Fuel Oil Testing Program, FPL will need to determine if the proposed program change is at least as, if not more, effective, in detecting unsatisfactory fuel oil. Since the proposed changes do not involve a change in the design of any plant system or component, and since the proposed changes will need to evaluate the effect of any ASTM standard edition change on the level of EDG reliability, the change proposed will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The proposed changes to the Technical Specifications will permit the Technical Specification required testing of Emergency Diesel Generator (EDG) fuel oil using more recent editions of the American Society for Testing and Materials (ASTM) standards currently listed in Technical Specification Surveillance Requirements 4.8.1.1.2e. and 4.8.1.1.2f. Prior to changing the edition of the previously approved ASTM standard being used to evaluate the EDG fuel oil, the proposed edition standard will be evaluated pursuant to 10 CFR Sec. 50.59, ``Changes, tests, and experiments.'' Title 10 CFR Sec. 50.59 permits a licensee to make changes in the procedures as described in the safety analysis report without prior NRC approval, provided that the proposed changes does not involve an unreviewed safety question. Title 10 CFR Sec. 50.59(a)(2) states that a proposed change involves an unreviewed safety question (iii) if the margin of safety as defined in the basis for any technical specification is reduced. Consequently, since any change to the edition of the ASTM standard to be used to evaluate EDG fuel oil acceptability must be evaluated relative to the more restrictive evaluation criterion of 10 CFR Sec. 50.59, then operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

ACTION statement g. of TS 3.8.1.1 is added to address the required action in the event the new fuel oil properties do not meet the Diesel Fuel Oil Testing Program limits. A failure to meet the API gravity, kinematic viscosity, flash point or clarity limits is cause for rejecting the new fuel oil prior to the addition to the Diesel Fuel Oil
Storage Tanks, but does not represent a failure to meet the Limiting

Condition for Operation (LCO) of TS 3.8.1.1, since the new fuel oil has
not been added to the storage tanks. Provided these new fuel oil properties are met subsequent to the addition of the new fuel oil to the storage tanks, 30 days is provided to complete the analyses of the other fuel oil properties specified in Table 1 of ASTM-D975-81, except sulfur which may be performed in accordance with ASTM-D1552-79 or ASTM-D2622-82. In the event the other new fuel oil properties specified in Table 1 of ASTM-D975-81 are not met, ACTION statement g. of TS 3.8.1.1 provides an additional 30 days to meet the Diesel Fuel Oil Testing Program limits. This additional 30 day period is acceptable because the
fuel oil properties of interest, even if they are not within limits, would not have an immediate effect on EDG operation.

ACTION statement h. of TS 3.8.1.1 is added to address the required action in the event the stored fuel oil total particulates does not meet the Diesel Fuel Oil Testing Program limits. Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel
oil will not burn properly in a diesel engine. The frequency for performing surveillance on stored fuel oil is based on stored fuel oil degradation trends which indicate that particulate concentration is unlikely to change significantly between surveillances.

Prior to changing the Turkey Point Units 3 and 4 Diesel Fuel Oil Testing Program, FPL will need to determine if the proposed program change is at least as, if not more, effective, in detecting unsatisfactory fuel oil. Since the proposed changes will require a safety evaluation to assure that the reliability of the EDGs using fuel
oil tested in accordance with the different ASTM standard edition maintains the current margin of safety, the proposed changes do not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Florida International University, University Park, Miami, Florida 33199.

Attorney for licensee: Harold F. Reis, Esquire, Newman and Holtzer,
P.C., 1615 L Street, NW., Washington, DC 20036.

NRC Project Director: Mohan C. Thadani, Acting.

Florida Power Corporation, et al.

Docket **No**. 50-302

Crystal River Nuclear Generating Plant, Unit **No**. 3, Citrus County, Florida.

Date of amendment request: September 30, 1994.

Description of amendment request: The proposed amendment would revise the Crystal River 3 (CR3) Nuclear generating Plant Technical Specifications (TS) to allow an increase in the rated thermal power (RTP) for CR-3 from the current 2544 level to 2568 Megawatt thermal (Wt). Accordingly, in TS 1.1, ``Definitions,'' would be revised to indicate the new power level of 2568 MWt. The proposed change would not require any hardware modifications.

Basis for proposed **no** significant hazards consideration determination: Currently, CR-3 is operating at a maximum RTP of 2544 MWt. The licensee proposes to operate at a maximum RTP of 2568 MWt, an increase of 24 MWt over the current licensed power of 2544 MWt.

The licensee states that the Babcock and Wilcox (B&W) 177 Fuel Assembly (FA) Nuclear Steam Supply System (NSSS) in the CR3 design is capable of operating at a thermal power level of 2772 MWt. Due to limitations in the secondary area of the plant, the licensee requests authorization to operate at 2568 MWt which is less than the design level of 2772 MWt. The licensee performed a detailed engineering study on this power increase.

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

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated. The thermal-hydraulic and nuclear characteristics of the reactor core were originally designed for a rated thermal power of 2568 MWt or higher. Therefore, the proposed thermal power increase to the reference power level of 2568 MWt does not change the original design assumptions and analyses for the reactor core. Most of the design basis accidents and transients were originally evaluated at the proposed power level. As described more fully in this submittal, those transients and accidents that were not originally evaluated at 2568 MWt were re-evaluated using CR-3 FSAR [Final Safety Analysis Report] Chapter 14 accident sequence

of events, reactor protection criteria, and approved calculational methods. Based on this evaluation and initial plant design evaluations,
FPC [Florida Power Corporation, the licensee for CR3] has determined that the probability and consequences of design basis transients and accidents are not significantly increased and that the radiological consequences from the design basis transients and accidents remain well
below 10 CFR 100 limits.

FPC has also reviewed CR-3 balance of plant and safety related systems to determine which systems and components could be affected by the proposed power increase. The changes to the reactor coolant system and secondary conditions and parameters are discussed in this submittal. These changes are minor in nature. The only Technical Specification change is to revise the reference power to 2568 MWt. **No** facility modifications will be required. FPC evaluated the systems and components and concluded that these systems and components will continue to perform within their design parameters with the unit operating at 2568 MWt.

Based on the foregoing, the proposed amendment does not significantly increase the probability or consequences of an accident previously evaluated.

2. The proposed thermal power increase does not create the possibility of a new or different kind of accident from previously evaluated accidents. As noted above, the thermal-hydraulic and nuclear characteristics of the reactor core were originally designed for operation at the proposed thermal power. Therefore, operation at the proposed power level does not introduce new or different performance characteristics that create the possibility of a new or different kind of accident.

FPC has also reviewed CR-3 safety-related systems and balance of plant systems to determine which systems could be affected by the proposed power increase and the resultant minor changes in plant parameters and operating conditions. Systems that could be affected were evaluated using the licensing basis criteria described in the CR-3
FSAR to assure their adequacy at the increased power level. Included in
these evaluations were plant features that are not power level related or directly affected by an increase in power level, as well as, associated issues such as environmental considerations. Equipment performance and plant operation were evaluated with respect to actual performance versus projected operating conditions to identify any

hardware modifications required to achieve the upgraded power. Based on
this evaluation, FPC has determined that all systems will continue to perform within their design parameters at 2568 MWt and that **no** physical
modifications to these systems will be necessary to accommodate a 2568 MWt rating. Only minor re-calibration of plant instrumentation to reflect the increased power will be needed. The proposed power level does not introduce any new performance characteristics or modes of operation for plant systems and components, and does not introduce any new failure modes.

Based on the foregoing, the proposed amendment does not create the possibility of a new or different kind of accident.

3. The proposed amendment does not involve a significant reduction in a margin of safety. The thermal-hydraulic and nuclear characteristics of the reactor core were originally designed for operation at the proposed power level. Most of the design basis transients and accidents were originally analyzed assuming a power level of 2568 MWt or higher. As described more fully in this submittal,
those transients and accidents that were not originally analyzed at 2568 MWt were re-evaluated using CR-3 FSAR Chapter 14 accident sequence
of events, reactor protection criteria, and approved calculational methods. FPC has determined that operation with the proposed thermal power will be bounded by the original analyses. In addition, FPC's evaluation of affected plant systems and components revealed that plant
systems and components will continue to operate within their design parameters with **no** significant change in a margin of safety.

Based on the foregoing, the proposed amendment does not involve a 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: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 32629

Attorney for licensee: A. H. Stephens, General Counsel, Florida Power Corporation, MAC-A5D, P. O. Box 14042, St. Petersburg, Florida 33733.

NRC Project Director: Mohan C. Thadani, (Acting).

Indiana Michigan Power Company

Docket Nos. 50-315 and 50-316

Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan.

Date of amendment request: August 3, 1994.

Description of amendment requests: The proposed amendments would allow the radiological effluent technical specifications (TS) to be relocated to other controlled documents. Procedural details contained in the current radiological effluents TS have been relocated to either the Offsite Dose Calculation Manual (OCDM) or the Process Control Program (PCP), as applicable. Proposed revisions to the OCDM and PCP have been prepared in accordance with the proposed changes to the administrative controls section of 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:

Criterion 1

The changes described above in **no** way negatively impact the requirements of the T/Ss. Separating the turbine room sump releases from the others is purely a clarification of the method we handle releases. The six ground monitoring wells added to the T/S table updates our current monitoring practice. With the six extra wells to monitor, we exceed the monitoring requirements of the T/Ss. Therefore, it is concluded that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2

No changes to the LCOs for either T/S are proposed as part of this amendment request. The proposed change does not involve any physical changes to the plant or any changes to plant operations. The changes merely propose to update our methods of implementing the T/S with our current practices. Thus, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3

The changes described above in **no** way negatively impact the requirements of the T/Ss. Separating the turbine room sump releases from the others is purely a clarification of the method we handle releases. The six ground monitoring wells added to the T/S table updates our current monitoring practice. With the extra wells to monitor, we exceed the monitoring requirements called for in the T/Ss. Therefore, it is concluded that the proposed changes do not involve a 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 requests involve **no** significant hazards consideration.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085.

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

NRC Project Director: John N. Hannon.

Niagara Mohawk Power Corporation

Docket **No**. 50-410

Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York.

Date of amendment request: October 5, 1994.

Description of amendment request: The proposed license amendment would revise the applicability requirements of Technical Specification (TS) 3.7.3 to require operability of the Control Room Outdoor Air Special Filter Train System in Operational Conditions 1, 2, 3 and ** (when irradiated fuel is being handled in the reactor building and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel and uncovering irradiated fuel) rather than in all Operational Conditions and * * *. The applicability requirements for Action Statement b of TS 3.7.3 and for the Radiation Monitoring Instrumentation required operable by TS Tables 3.3.7.1-1 and 4.3.7.1-1 would be changed in a similar manner. The proposed amendment would also add a notation to Action Statement b.1 of TS 3.7.3 stating that the provisions of Specification 3.0.4 are not applicable provided an operable control room filter train is in the emergency pressurization mode of operation. The licensee stated that these proposed changes are consistent with the requirements of the NRC's Improved Standard Technical Specifications (NUREG-1433) and with Generic Letter 87-09, ``Section 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operation and Surveillance

Requirements.''

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

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.

The Control Room Outdoor Air Special Filter Train System is not an initiator or precursor to an accident. The Control Room Outdoor Air Special Filter Train System responds to a release of radioactivity to the outside environment as detected in the air supply to the control room by providing a radiologically controlled environment within the control room. In operational conditions 4 and 5, the probability and consequences of a design basis accident are reduced due to the pressure and temperature limitations in these operational conditions. Therefore, maintaining the chiller subsystem operable is not required in operational conditions 4 and 5, except for the * * * operational condition. Therefore, a change to applicability and action statements of LCO [Limiting Condition For Operation] 3.7.3 cannot affect the probability of a previously evaluated accident.

All accidents which take credit for operation of the Control Room Outdoor Air Special Filter Train System in the emergency pressurization mode of operation are analyzed and presented in Chapter 15 of the USAR [Updated Safety Analysis Report]. These accidents can only occur in operational conditions 1, 2, 3 and * * *.

Accordingly, the proposed change in the applicability of LCO 3.7.3 from all operational conditions (i.e., 1, 2, 3, 4, 5 and * * *) to operational conditions 1, 2, 3 and * * * does not significantly increase the consequences of an accident previously evaluated. The proposed change to action statement b of LCO 3.7.3 and to Tables 3.3.7.1-1 and 4.3.7.1-1 of LCO 3.3.7.1 is consistent with the above change.

Sections 15.7.4 and 15.7.5 of the USAR evaluate a fuel handling accident and a spent fuel cask drop accident, respectively. The radiological evaluation of these accidents considers the unfiltered radioactivity that enters the control room prior to the automatic operation of the Control Room Outdoor Special Filter Train System in the emergency pressurization mode of operation. The radiological consequences of these accidents are within the limits of GDC [General

Design Criterion]-19.

With one control room filter train inoperable and prior to entering the operational condition, the proposed change to action statement b.1 of LCO 3.7.3 would require an operable control room filter train be placed in the emergency pressurization mode of operation. During an accident involving the release of radioactivity to the environment, an operable control room filter train would already be running in the emergency pressurization mode and performing its safety function, thereby preventing the entry of unfiltered radioactivity into the control room. Therefore, if a fuel handling accident or a spent fuel cask drop accident were to occur and release radioactivity, the control room personnel radiological doses would be less than the doses depicted in the USAR. Accordingly, the Technical Specification change to action statement b.1 does not significantly increase the consequences of a previously evaluated accident.

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.

This amendment does not involve any accident precursors or initiators. In addition, this amendment does not require any changes to plant equipment.

During an accident involving the release of radioactivity to the environment an operable control room filter train would already be running in the emergency pressurization mode and performing its safety function. Furthermore, the operating status of a running control room filter train would be unaffected by the receipt of an automatic start signal due to high radiation in either air intake to the Control Room Outdoor Air Special Filter Train System. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any 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.

The proposed change in the applicability of LCO 3.7.3 from all operational conditions (i.e., 1, 2, 3, 4, 5 and * * *) to operational conditions 1, 2, 3 and * * * is consistent with the safety analysis contained in the USAR. The proposed changes to action statement b of LCO 3.7.3 and to Tables 3.3.7.1-1 and 4.3.7.1-1 of LCO 3.3.7.1 is

consistent with the above change.
    Entry into the ** operational condition for LCO 3.7.3 with one control room filter train inoperable and the other control room filter train operable and operating in the emergency pressurization mode provides a comparable level of safety to two operable non-running control room filter trains. The remedial measure prescribed by Technical Specification action statement b.1 (placing an operable control room filter train in the emergency pressurization mode of operation) for which the exception to LCO 3.0.4 is proposed provides a sufficient level of protection to permit operational mode changes and safe long-term operation of NMP2 [Nine Mile Point Unit 2] consistent with the licensing basis described in the USAR. Therefore, the proposed change to action statement b.1 is consistent with Generic Letter 87-09, ``Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operation and Surveillance Requirements.'' Accordingly, this change will not significantly reduce the margin of safety.
    This proposed amendment is consistent with the Improved Standard Technical Specifications, NUREG-1433. Accordingly, as determined by the analysis above, this proposed amendment involves **no** significant hazards consideration.
    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: Ledyard B. Marsh.

Niagara Mohawk Power Corporation

Docket **No**. 50-410

    Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York.
    Date of amendment request: October 21, 1994.

Description of amendment request: The proposed amendment would add a footnote to Technical Specification (TS) 4.8.1.1.2.e.8 which would permit performance of the 24-hour functional test of the emergency diesel generators (EDGs) during power operation. TS 4.8.1.1.2.e.8 currently requires the 24-hour functional test of the EDGs be performed at least once per 18 months during shutdown; the proposed amendment would permit this testing to be performed during power operation provided the other two EDGs are operable. If either of the other two EDGs become inoperable, the test would be aborted.

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.

The proposed change to permit the 24 hour functional test of the diesels to be performed during power operation does not increase the chances for a previously analyzed accident to occur. The function of the diesels is to supply emergency power in the event of a loss of offsite power. Operation of the diesels is not a precursor to any accident. Furthermore, the diesel generator being tested will remain operable and will be available to supply emergency loads within the required time. In addition, the two remaining diesel generators will be operable during the test. Consequently, if an offsite disturbance were to occur that affected the operability of the diesel being tested, the two remaining diesels would be capable of feeding the loads necessary for safe shutdown of the plant. This addresses the concerns raised in Information Notice 84-69 regarding the operation of emergency diesel generators connected in parallel with offsite power. In summary, the proposed changes do not adversely affect the performance or the ability of the diesel generators to perform their intended function.

Therefore, the proposed change 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 previously evaluated.

The proposed amendment to the 24 hour functional surveillance test will not affect the operation of any safety system or alter its response to any previously analyzed accident. The diesel will automatically transfer from the test mode if necessary to supply emergency loads in the requried time. The test mode is used for the monthly surveillance of the diesel generators as well, therefore, **no** new plant operating modes are introduced. In the event the diesel fails the functional test it will be declared inoperable and the actions required for an inoperable diesel will be performed. The remaining two diesel generators will be operable and are capable of feeding the loads necessary for safe shutdown of the plant.

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

The proposed amendment will not reduce availability of the diesel generator being tested to provide emergency power in the event of a loss of offsite power. If a loss of offsite power or a loss of coolant accident occurs during the surveillance test, the emergency bus would de-energize and shed load. The diesel generator would then transfer from the test mode to the emergency mode. It would then be available to automatically supply emergency loads. In addition, the two remaining generators will be operable during the test. Consequently, if an offsite disturbance were to occur that affected the operability of the diesel begin tested, the two remaining diesels would be capable of feeding the loads necessary for safe shutdown of the plant. The time required for the diesel being tested to pick up emergency loads will not be affected by performing the 24 hour functional test during power operation.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Ledyard B. Marsh.

Northeast Nuclear Energy Company et al.

Docket **No**. 50-336.

Millstone Nuclear Power Station, Unit **No**. 2, New London County, Connecticut.

Date of amendment request: October 18, 1994.

Description of amendment request: The proposed amendment would require three type A overall Integrated Containment Leakage Tests be conducted at approximately equal intervals during shutdowns during each 10-year service period. For the third Type A test for the second 10-year period, it would be conducted during the thirteenth refueling outage extending the second 10-year service period to the end of the thirteenth refueling outage. The amendment would also change the Containment Leakage Bases by reflecting the conditions of a proposed exemption to 10 CFR 50, Appendix J, that would remove the requirement that the third Type A test for each 10-year period be conducted when the plant is shutdown for the 10-year plant inservice inspection.

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

* * * The proposed changes do not involve a SHC [significant hazards consideration] because the change would not:

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

Type A tests are performed to ensure that the total leakage from containment does not exceed the maximum allowable primary containment leakage rate at the design pressure. This ensures compliance with the dose limits of 10 CFR 100.

The proposal to revise Surveillance Requirement 4.6.1.2.a of the Millstone Unit **No**. 2 Technical Specifications will increase the flexibility for scheduling the Type A tests. It does not modify the maximum allowable leakage rate at the design containment pressure, does not impact the design basis of the containment, and does not make any physical or operational changes to existing plant structures, systems, or components.

The first two Type A tests of the second 10-year service period

for
Millstone Unit **No**. 2 have been conducted. The results of these tests demonstrate that Millstone Unit **No**. 2 has maintained control of containment integrity by maintaining margin between the acceptance criterion and the ``As-Found'' and ``As-Left'' leakage rates.

Historically, Type A tests have a relatively low failure rate where
Type B and C testing (local leakage rate tests) could not detect the leakage path. Most Type A test failures are attributed to failures to Type B or C components (containment penetrations and isolation valves).
Type B and C components are tested per Surveillance Requirement 4.6.1.2.d for the Millstone Unit **No**. 2 Technical Specifications. These tests are required to be conducted at intervals **no** greater than 24 months, and the acceptance criterion for the combined leakage rate for all penetrations and valves subject to the Type B and C tests is 0.6 L<INF>a. These local leakage rate tests provide assurance that containment integrity is maintained. The relatively low ``As-Left'' Type B and C total leakage resulting from the past outage indicates that the leakage has been maintained within the technical specification
acceptance criterion. The Type B and C tests will continue to be performed in accordance with the requirements of Surveillance Requirement 4.6.1.2.d. However, on September 26, 1994, NNECO submitted a request for a one-time technical specification change, request for enforcement discretion, and a request for a scheduler exemption from Appendix J to 10 CFR 50 regarding the Schedule for Type B and C testing. The NRC verbally granted enforcement discretion on September 24, 1994, and written enforcement discretion on September 30, 1994. The
schedular exemption request was granted on October 12, 1994.

The previous Type A, B, and C tests demonstrate that Millstone Unit
No. 2 has maintained control of containment integrity by maintaining a conservative margin between the acceptance criterion and the ``As-Found'' and ``As-Left'' leakage results. Based on this, the Millstone Unit **No**. 2 containment is considered to be in sound condition. **No** operations are known to have occurred which would suggest any substantial degradation of these results.

Based on the above, the proposal to revise Surveillance Requirement
4.6.1.2.a of the Millstone Unit **No**. 2 Technical Specifications does not

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

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

The proposal to revise Surveillance Requirement 4.6.1.2.a of the Millstone Unit **No**. 2 Technical Specifications will increase the flexibility in scheduling the Type A tests. It does not make any physical or operational changes to existing plant structures, systems, or components. In addition, the proposal does not modify the acceptance
criterion for the Type A tests. Maintaining the leakage through the containment boundary to the atmosphere within a specific value ensures that the plant complies with the requirements of 10 CFR 100. The containment boundary serves as an accident mitigator; it is not an accident initiator. Therefore, the proposal to revise Surveillance Requirement 4.6.1.2.a does not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposal to revise Surveillance Requirement 4.6.1.2.a of the Millstone Unit **No**. 2 Technical Specifications will increase the flexibility for scheduling the Type A tests. It does not modify the maximum allowable leakage rate at the design containment pressure, does
not impact the design basis of the containment, and does not make any physical or operational changes to existing plant structures, systems, or components.

The first two Type A tests of the second 10-year service period for
Millstone Unit **No**. 2 have been conducted. The results of these tests demonstrate that Millstone Unit **No**. 2 has maintained control of containment integrity by maintaining margin between the acceptance criterion and the ``As-Found'' and ``As-Left'' leakage rates. Additionally, the results of the last Type B and C tests had significant margin with respect to the acceptance criterion. Based on the previous Type A, B, and C tests, the Millstone Unit **No**. 2 containment is considered to be in sound condition. **No** operations are known to have occurred which would suggest any substantial degradation of these results.

Based on the above, the proposal 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: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New
London Turnpike, Norwich, CT 06360.

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

NRC Project Director: Phillip F. McKee.

Northeast Nuclear Energy Company, et al.

Docket **No**. 50-423

Millstone Nuclear Power Station, Unit **No**. 3, New London County, Connecticut.

Date of amendment request: September 30, 1994.

Description of amendment request: The licensee has proposed to revise the Technical Specifications (1) to clarify the definition of core alterations, (2) to change the verbiage in the Limiting Condition For Operation (LCO) addressing the refueling operations, (3) to make changes to three surveillance requirements involving source range instrumentation, and (4) to change the LCO regarding the Residual heat Removal and coolant circulation water levels to be consistent with the guidance provided in NUREG-1431.

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

The proposed changes do not involve an SHC [significant hazards consideration] because the changes would not:

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

Boron Dilution in Mode 6--A boron dilution in Mode 6 is precluded by technical specification requirements to close and lock all dilution source valves. There is a provision for dilution valves to be opened under administrative controls; in this case, cautionary measures will be taken to control and monitor the reactivity addition. Deletion of the source range analog operational test prior to core alterations will
not impact an accident previously evaluated since the sources range

monitors are verified operable prior to entry into Mode 6 and every 7 days thereafter. The change in definition for a core alteration means that components which do not effect reactivity may be moved within the reactor vessel without any additional condition such as direct supervision of an SRO.

Since a boron dilution would not be initiated by movement of nonfuel components within the reactor vessel, it is not impacted by the change in definition of a core alteration.

Inadvertent Loading of a Fuel Assembly--Movement of a fuel assembly would be performed as a core alteration under the supervision of an SRO, therefore, it would not be impacted by the change to the definition of a core alteration. The change to the source range monitors also will not affect the probability of occurrence of a misloaded fuel assembly since this accident is precluded by administrative controls, as well as the source range monitors. Also, there will be **no** degradation in the reliability or accuracy of the source range monitors due to this change. The deletion of the requirement to perform the analog channel operational test within eight hours prior to core alterations will not impact performance of the monitors, since they have to be checked prior to entry into Mode 6 and every 7 days thereafter.

Fuel Handling Accident--Movement of fuel will not affect this accident, because it will still be considered a core alteration. Therefore, there is **no** effect on the probability of a fuel handling accident. The source range monitors are not involved in the occurrence of a fuel handling accident. The fuel handling accident is the only accident considered here with radiological consequences. It will not be impacted by the proposed changes.

Loss of RHR in Mode 6--The probability of this accident will not be changed since the new requirement is the same as before. As before, RHR may be secured for up to one hour per eight-hour period and boron dilution operations may not be performed with RHR secured (although this requirement is being added to the notes, the requirement is also given elsewhere in the technical specifications). Additionally, the existing reactor coolant system (RSC) temperature limits must still be met.

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

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

All required systems will continue to operate as before. Therefore, there is **no** possibility of a new or different kind of accident. The deletion of the source range analog channel operational test within eight hours prior to core alterations will not affect the performance of the monitors since they will have had this test completed prior to entry into Mode 6 and every 7 days thereafter. The change in definition of a core alteration cannot create the possibility of a new type of accident because those initiating events for accidents will remain classified as core alterations.

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

The margin of safety for the above listed accidents will remain as before.

a. Boron dilution in Mode 6--This accident calculates the time from receipt of a shutdown margin monitor dilution alarm until the core reaches criticality. Since this time is not changed, there is **no** reduction in the margin of safety. In this case, the dilution is precluded by administrative controls which will not be impacted by the proposed changes.

b. Inadvertent Loading of a Fuel Assembly--Technical Specification 3.9.1.1 protects against this accident by requiring sufficient boron in the RCS to prevent criticality for any core configuration including two stuck RCCAs [rod cluster control assemblies] in the fully withdrawn position. Since this requirement will not change, the margin of safety will not change.

c. Fuel Handling Accident--The margin of safety for the radiological limits is not changed.

d. Loss of RHR--Changes are editorial due to the revised definition of a core alteration. There is **no** change to the margin of safety.

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

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

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

NRC Project Director: Phillip F. McKee.

Northern States Power Company

Docket Nos. 50-282 and 50-306

Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, Goodhue County, Minnesota.

Date of amendment requests: October 3, 1994.

Description of amendment requests: The proposed amendment would revise Prairie island Nuclear Generating Plant Technical Specification 4.6, ``Periodic Testing of Emergency Power Systems.'' Specifically, the proposed amendment would modify the emergency diesel generator (EDG) 24-hour load test requirements to provide a indicated load range of 103-110% of the continuous rating. The proposed amendment would also rephrase various EDG test requirements to provide clarity and delete the requirement to verify that the auto-connected loads do not exceed 3000 kw (Unit 2 5100kw).

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

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

Changing the specification from ``unit'' to ``diesel generator'' does not change the intent of the specification, it merely clarifies the original intent and therefore cannot involve a change in the probability or consequences of an accident.

Changing the 22-hour lower range limit from a load of 90% to an indicated load of 92% removes possible ambiguity from the specification but does not change the actual requirement, therefore it cannot involve a change in the probability or consequences of an accident.

Removing the 22-hour upper range limit from the specification does not reduce the conservatism of the test since operating at a higher load provides more evidence of the ability of the machine to carry the accident loads. For this reason, this change will not involve any increase in the consequences of an accident. Also, increasing the load at which the diesel generator is tested cannot affect the probability of an accident.

The NRC staff has pointed out, in Generic Letter 88-15, the hazards of testing the Diesel Generator at a load greater than the design rating. The proposed change is intended to ensure that the design rating is not inadvertently exceeded. Since the recent installation of two additional emergency diesel generators, the highest anticipated event loads are: Unit 1-2414kW, Unit 2-3813 kW. For these diesel generators, then, 103% of the continuous ratings:

<bullet> Unit 1, 103% of 2750 kW (continuous rating) = 2832.5 kW represents 117.3% of the highest anticipated event load and;

<bullet> Unit 2, 103% of 5400 kW (continuous rating) = 5562 kW represents 145.9% of the highest anticipated event load.

A test load of 103%, therefore would still be significantly greater than the load required during accident conditions. Since an adequate level of electrical load carrying capacity of the diesel generators (and thus their accident mitigating functions) would still be demonstrated by the surveillance test, the consequences of an accident would be unaffected by the proposed change. The probability of occurrence of a previously evaluated accident would be unaffected since testing a diesel generator at load between 103 and 110 percent instead of at load between 105 and 110 percent could not cause or contribute to the initiation of an accident. For these reasons, this change could have **no** effect on the probability or consequences of an accident previously evaluated.

Allowing momentary transients outside of the test band does not affect the conduct of the test, it merely allows momentary swing outside the specified band to not invalidate the test. Not allowing momentary transients would not prevent them, it would only require conducting the test longer until the specified time period was achieved without moving outside the band. Since the machine will not be operated any differently, this specification change cannot affect the

probability or consequences of an accident previously evaluated.
    Proposed changes A, B, C, D, and the first part of E [identified as
such in the submittal] are intended to clarify the meaning of the existing specifications without changing the requirements. For this reason, these proposed changes to the Technical Specifications will not
change the manner in which the plant is operated or maintained. These administrative changes, therefore, will effect on the probability or consequences of an accident previously evaluated.
    The second part of E (verification of the bypass of diesel generator trips during a simulated safety injection signal vs concurrent safety injection and loss of offsite power signals) does not
change the intended function which is to be tested but, rather, reduces
the special conditions (temporary electrical jumpers to simulate the loss of offsite power) in which the plant needs to be placed in order to perform the test.
    Proposed change F (removal of the verification that the auto-connected load do not exceed 3000 or 5100 kW) does not reduce the assurance of the ability of the diesel generators to perform the accident mitigation functions since this verification is performed by other, more pertinent, means.
    Therefore, these changes cannot increase the probability or consequences of an accident previously evaluated.
    2. The proposed amendment will not create the possibility of a new or different king of accident from any accident previously analyzed.
    Changing the specification from ``unit'' to ``diesel generator'' does not change the intent of the specification, it merely clarifies the original intent and therefore cannot create the possibility of a new or different kind of accident.
    Changing the 22-hour lower range limit from a load of 90% to an indicated load of 92% removes possible ambiguity from the specification
but does not change the actual requirement.
    Removing the 22-hour upper range limit from the specification does not change the manner in which the surveillance is performed. It only affects whether the time spent above 100% load can be counted toward 22
hours in the 22-hour portion of the test. This change would not allow any new modes of operation nor does it allow any modification to the plant.

As stated above, testing a diesel generator at a load between 103 and 110% instead of between 105 and 110% could not cause or contribute to the initiation of an accident.

Allowing momentary transients outside of the test band does not affect the conduct of the test, it merely allows momentary swings outside the specified band to not invalidate the test. Not allowing momentary transients would not prevent them, it would only require conducting the test longer until the specific time period was achieved without moving outside the band.

Therefore, for these reasons, operation of the facility in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

As stated above [for changes A-F], the proposed changes will not cause a change in the way in which the plant is operated or maintained,
excepted for the reduction of the special conditions in which the plant
needs to be placed in order to test the bypass of the diesel generator trips. Therefore, these administrative changes will not create the possibility of a new or different kind of accident from any accident previously analyzed.

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

Changing the specification from ``unit'' to ``diesel generator'' does not change the intent of the specification, it merely clarifies the original intent and therefore cannot affect the margin of safety.

Changing the 22-hour lower range limit from a load of 90% to an indicated load of 92% removes possible ambiguity from the specification
but does not change the actual requirement and therefore cannot affect the margin of safety.

The margin of safety is not affected by removal of the 22-hour upper range limit on the operation of the diesel generators during surveillance testing since the margin of safety is related to the magnitude of the accident loads and the maximum capacity of the machine
to carry load and this margin would be unaffected by this change.

The capacity of each diesel generator to carry electrical load can not be diminished by being tested at a lower load. Also, load testing to less than 105% but more than 103% does not lessen the confidence in the ability of the diesel generators to carry adequate load for this facility since these diesel generators have significantly greater load capacity than required by Standard Review Plan guidance in this regard (the guidance allows peak accident load up to 100% of the continuous rating versus Unit 1 diesel generators peak accident load of 87.8% and Unit 2 diesel generators peak accident load of 70.6%). Therefore, this change will not involve a significant reduction in the margin of safety.

Allowing momentary transients outside of the test band does not affect the conduct of the test, it merely allows momentary swings outside the specified band to not invalidate the test. Not allowing momentary transients would not prevent them, it would only require conducting the test longer until the specified time period was achieved without moving outside the band. Since the machine will not be operated any differently per the new specification, the margin of safety is unaffected.

As stated above [for changes A-F], the proposed changes will not cause a change in the way in which the plant is operated or maintained, except for the reduction of the special conditions in which the plant needs to be placed in order to test the bypass of the diesel generator trips. Therefore, these administrative change will not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John N. Hannon.

Omaha Public Power District

Docket **No**. 50-285

Fort Calhoun Station, Unit **No**. 1, Washington County, Nebraska.

Date of amendment request: October 7, 1994.

Description of amendment request: The proposed amendment to the

Technical Specifications (TSs) would (1) delete the surveillance requirements contained in TS 3.6(3)a for the raw water backup valves to
the containment cooling coils, (2) delete the surveillance requirements
contained in TS 3.2, Table 3-5, item 6, for raw water valves, and (3) revise the basis of TS 2.4 to reflect these changes.

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

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

The deletion of surveillance requirements contained in Technical Specifications (TS) 3.2, Table 3-5, Items 6 and 3.6(3)a does not involve a significant increase in the probability or consequences of an
accident previously evaluated.

TS 3.6(3)a requires the Raw Water (RW) backup valves to the containment air coolers to be tested each refueling outage. In 1990, during the process of reviewing several open items created by the design basis reconstitution project, an engineering analysis determined
that RW direct cooling of the containment air cooling coils should not be used after an accident that has created elevated temperature conditions inside containment. The high containment air temperatures, in conjunction with the low back pressure in the containment cooling coils when in the RW direct cooling mode, introduces the possibility of
vaporization inside the coils. Therefore, the use of RW direct cooling for the containment air coolers has been discontinued in post-Loss of Coolant Accident (LOCA) or post-Main Steam Line Break (MSLB) situations. The issue of not being able to utilize RW direct cooling to
the containment air cooling coils was reported to the NRC in LER-90-25,
dated October 29, 1990 and LER-90-25 Revision 1, dated December 17, 1990.

Raw water direct cooling of the containment air coolers is possible
if the containment atmospheric temperatures are less that 150 deg.F. If
RW direct cooling of the containment air coolers was utilized after a

LOCA or MSLB accident, it could only be used for long-term containment atmospheric cooling. These conditions are essentially equivalent to that associated with conditions in containment during normal plant operation. RW direct cooling of the containment air coolers is not a required post-accident function to maintain containment pressure below 60 psig. Since these valves are not required to perform a post-accident function, deletion of the requirements to test these valves does not involve a significant increase in the probability or consequences of an accident previously evaluated.

TS 3.2, Table 3-5, Item 6 requires that valves in the RW system be tested every refueling outage. The valves tested by this surveillance that could perform a safety function are already tested in accordance with TS 3.3(1). Therefore testing of these valves under TS 3.2, Table 3-5, Item 6 is redundant to TS 3.3(1)a.

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

There will be **no** physical alterations to the plant configuration, changes to setpoint values, or changes to the implementation of setpoints or limits as a result of this proposed change. Valves that are required to be repositioned during an accident to mitigate the consequences will still be tested on a refueling frequency. The proposed change only deletes unnecessary or redundant testing requirements from the TS. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed changes delete unnecessary or redundant surveillance requirements within the TS. The deletion of TS 3.2, Table 3-5 Item 6, only deletes testing requirements that are already required to be conducted by TS 3.3(1)a. The deletion of the requirement to test the RW backup valves to the containment air coolers in TS 3.6(3) only deletes an unnecessary surveillance. RW direct cooling of the containment air coolers is not required to maintain containment pressure below the design limit of 60 psig. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Theodore R. Quay.

Pennsylvania Power and Light Company

Docket Nos. 50-387 and 50-388

Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania.

Date of amendment request September 26, 1994.

Description of amendment request: The amendment would remove the requirement for operability of the Average Power Range Monitors (APRMs) while the plant is in Operational Condition 5. However, the requirement for the APRMs to be operable during a shutdown margin demonstration, when the mode switch is in Startup, will remain unchanged.

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 construction, which is presented below:

I. This proposal does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Not requiring APRMs to be OPERABLE in OPCON 5 will not increase the probability of inadvertent reactor critically during refueling operations. Refueling Interlocks, NMS [Neutron Monitoring System] (SRMs [Source Range Monitor], IRMs [Intermediate Range Monitor]), and procedural restrictions provide assurance that inadvertent criticality does not occur due to the simultaneous withdrawal or removal of two control rods or due to the inadvertent insertion of a fuel bundle into a core location with a control blade removed.

The FSAR [Final Safety Analysis Report] Section 15.4.1 discusses the potential for a control rod withdrawal error during refueling and start-up operations. The discussion concludes that the withdrawal of one control rod does not require a safety action because the total worth of one control rod is not sufficient to cause criticality. The attempted withdrawal of two control rods, assuming an operator error and a single active failure, would result in a control rod block initiated by the Refueling Interlocks. The safety-related IRM subsystem, which is required by Technical Specifications to be OPERABLE while in OPCON 5, is designed to generate a rod block or reactor scram on high neutron flux and is therefore a backup protective system for the Refueling Interlocks during refueling.

The Safety-related IRM subsystem of the NMS is required by Technical Specifications to be OPERABLE during OPCON 5 to support the safety design bases of the NMS and RPS [Reactor Protection System]. The SRM is not a safety-related subsystem but is important to plant safety and is required by Technical Specifications to be OPERABLE in OPCON 5. The SRM subsystem provides the plant operator with neutron flux levels from startup conditions to the IRM operating range. The SRMs and IRMs are designed to respond to local core conditions and would indicate and respond (control rod block or scram) to an accident condition to mitigate the transient. Thus, the APRMS are not necessary to be OPERATOR in OPCON 5. The proposed Technical Specification change will not alter the current requirements that the APRMs be OPERABLE during shutdown margin demonstrations in OPCON 5 when the mode switch is in Startup.

The proposed Technical Specification change would reduce the APRM operability requirement in OPCON 5 and would not affect the FSAR evaluation of the inadvertent criticality due to the withdrawal or removal of the highest worth control rod or due to the insertion of fuel bundles in uncontrolled cells. The FSAR concludes that the Refueling Interlocks and plant procedures provide assurance that inadvertent criticality does not occur during refueling.

The consequences of an accident will not be increased by the proposed Technical Specification change because of the existing lines of defense which prevent an inadvertent criticality event during refueling, e.g., administrative restrictions, refueling procedures, licensed plant operators, SRMs, Refueling Interlocks, and IRMs. Furthermore, should the number of operator IRM or SRM channels be less than that required by Technical Specifications, the Technical Specifications require that core alteration activities be suspended and all insertable control rods be inserted into the core.

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

II. This proposal does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to the Technical Specifications will remove the APRM operability requirement while in OPCON 5 (except for shutdown margin demonstration testing); however, the SRMs and IRMs will still be required to be OPERABLE in OPCON 5.

The IRMs are safety-related and are designed to detect and respond to increases in neutron flux within the local core regions. Any inadvertent increases in neutron flux during refueling would originate at a local core location, i.e., rod withdrawal error or fuel bundle insertion. Technical Specifications require IRM operability and will generate an RPS scram or control rod block if neutron flux increased to the setpoint. Therefore, removing the APRMs operability requirement in OPCON 5 would not effect any safety related equipment or equipment important to safety.

The APRMs provide core power information to the control room operator and also provide trip signals to the RMCS [Reactor Manual Control System] and RPS as required. The absence of an APRMs input signal will not affect these systems during refueling operations.

Removing the APRMs operability in OPCON 5 will not affect the response of safety-related equipment as previously evaluated in the FSAR. The proposed changes to the Technical Specifications do not affect any safety-related equipment or equipment important to safety.

The proposed changes to the Technical Specifications would remove the APRMs operability requirement during refueling operations. Technical Specifications require IRM operability and will generate an RPS scram or control rod block if neutron flux increased to the applicable setpoint.

No new types of accidents would be introduced since the SRMs and IRMs are available and required to be OPERABLE in OPCON 5. Both SRMs and IRMs would indicate and provide a control rod block or scram signal, as appropriate, to an increase in neutron flux to mitigate a transient event. Furthermore, should the number of OPERABLE IRM or SRM channels be less than that required by Technical Specifications, the Technical Specifications require that core alteration activities be suspended and all insertable control rods be inserted into the core.

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

III. This change does not involve a significant reduction in a

margin of safety.
For the reasons discussed in items 1 and 2 above and because the Technical Specification Bases do not discuss or require APRMs operability during OPCON 5, Refueling, the proposed Technical Specification changes do not involve a significant reduction in a margin of safety.
The NRS staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves **no** significant hazards consideration. Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701 Attorney for licensee: Jay Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street NW., Washington, DC 20037.
NRC Project Director: John F. Stolz.

Philadephia Electric Company

Docket Nos. 50-352 and 50-353

Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania.
Date of amendment request: August 22, 1994.
Description of amendment request: The amendment consists of five (5) sections of Technical Specifications changes which reflect the Improved Standard Technical Specifications (NUREG-1433):

Section 1: Control Rod Block Instrumentation,
Section 2: Standby Liquid Control System Operability in Mode 5,
Section 3: Scram Discharge Volume Valve Testing,
Section 4: Optional Method of Scram Timing, and
Section 5: Definition of Core Alteration.

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:
Section 1: Control Rod Block Instrumentation
1. The proposed Technical Specifications (TS) changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

    The proposed TS changes can be divided into two general categories,
the deletion of the ``S/U'' requirements, and the change in frequency
of the SRM [Source Range Monitor] and IRM [Intermediate Range Monitor]
Calibration and Functional Tests. In each case in which the ``S/U''
requirement has been deleted, the normal surveillance frequency
specified for the required Operating Condition remains. The equipment's
associated probability of failure remains unchanged. In the case of the
surveillance frequency changes proposed for the SRMs and IRMs, the
probability of an accident evaluated in the SAR [Safety Analysis
Report] occurring does not increase since there is **no** credit taken in
the SAR for those Control Rod Block functions with respect to an
accident. As such, the proposed changes will not result in an increase
in the probability of occurrence of an accident previously evaluated in
the SAR. The proposed TS changes do not alter the method of operation
or performance of the equipment in carrying out associated Control Rock
Block functions. Thus, the consequences of an accident previously
evaluated in the SAR are not increased.
    Therefore, the proposed TS changes do not involve an increase in
the probability or consequences of an accident previously evaluated.
    2. The proposed TS changes do not create the possibility of a new
or different kind of accident from any accident previously evaluated.
    The proposed TS changes do not alter the configuration of the plant
or the way that the plant is operated. The equipment can perform **no**
other function than it is presently capable of, or cause or permit any
other accident than is now possible. Thus, the possibility of an
accident of a different type than previously evaluated in the SAR
cannot be created.
    Therefore, the proposed TS changes do not create the possibility of
a new or different kind of accident from any previously evaluated.
    3. The proposed TS changes do not involve a significant reduction
in a margin of safety.
    Since the proposed TS changes affect only the surveillance
frequency intervals and do not change the plant configuration or
associated instrument setpoints, there is **no** quantitative or
qualitative reduction in the margin of safety. Thus, the margin of
safety as defined in the bases of any Technical Specification is not

reduced.
    Therefore, the proposed TS changes do not involve a reduction in a margin of safety.
Section 2: Standby Liquid Control System Operability in Mode 5
    1. The proposed Technical Specifications (TS) change does not involve a significant increase in the probability or consequences of an
accident previously evaluated.
    The proposed TS change will remove the SLCS operability requirement
in OPCON 5. The purpose of the SLCS is to bring the reactor to and maintain it in a cold shutdown condition from normal power operations following failure to scram during power operations. Initiation of the SLCS is not a precursor to any accident. Therefore, inoperability of the SLCS in OPCON 5 cannot increase the probability of an accident previously evaluated.
    The proposed TS change does not involve a physical change in any system's configuration and **no** new modes of operation are introduced. The SLCS has not analyzed function OPCON 5. The probability of fuel failure will not be increased by this change. Shutdown margin, in conjunction with TS requirements and procedural controls, will assure that an inadvertent criticality event will not occur during refueling. In addition, the Reactor Protection System (RPS) and Control Rod System
will provide protection in the unlikely event that an inadvertent criticality should occur.
    Therefore, the proposed TS change does not involve an increase in the probability or consequences of an accident previously evaluated.
    2. The proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
    The proposed TS changes does not involve a physical change in any system's configuration and **no** new modes of operation are introduced. The SLCS's only purpose is to mitigate the consequences of a failure to
scram during power operation. In OPCON 5, the SLCS has **no** analyzed function, therefore, the proposed TS change will not create the possibility of a new or different kind of accident from any previously evaluated.
    3. The proposed TS change does not involve a significant reduction in a margin of safety.
    The purpose of the SLCS is to bring the reactor to and maintain it in a cold shutdown condition from normal power operations following a failure to scram during power operations. The SLCS is not designed to

terminate an inadvertent criticality during OPCON 5. Shutdown margin, either demonstrated or analytically determined, in conjunction with Technical Specifications and procedural controls, will assure that an inadvertent criticality event will not occur during refueling operations. In addition, the RPS and Control Rod System, which are extremely reliable, will provide protection in the unlikely event that an inadvertent criticality does occur. Therefore, the proposed TS change does not involve a reduction in a margin of safety.

Section 3: Scram Discharge Volume Valve Testing

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

The Scram Discharge Volume (SDV) is not an accident initiator. Deletion of the requirement that the SDV be determined OPERABLE by testing the SDV vent and drain valves when control rods are scram tested from a normal control and configuration of less than or equal to 50% rod density at least once per 24 months, as proposed, will have **no** effect on the probability or consequences of an accident previously evaluated.

This proposed TS will have a negligible impact on the conditions experienced by the vent and drain valves as they stroke closed, since the SDV is initially vented to the atmosphere, and the valves close before the SDV becomes pressurized, even during a scram at full reactor power. Reactor pressure and Control Rod Drive (CRD) discharge flow conditions do not influence the SDV vent and drain closure rates, since the SDV is of sufficient volume and initially vented such that peak pressure prior to the SDV complete isolation will not be substantial. In addition, lower coolant temperatures expected during testing at shutdown conditions will also have a negligible impact on the performance of the test. Although, there could be some variation in the performance [of] the SDV vent and drain valves to re-open when performing the test during shutdown conditions, as opposed to conducting the test during power operation, the ability of the valves to re-open is demonstrated after each reactor scram during power operation.

In the event and SDV vent or drain valve failed to open, increasing SDV level during reactor operation would cause 1) an alarm in the Main

Control Room (MCR), 2) a control rod block, and finally a reactor scram
initiated by the Reactor Protection System (RPS) if action is not taken
to drain the SDV. Therefore, the ability to shut down the reactor is not impaired. If a SDV vent or drain valve fails to close, the redundant valve's closure would provide the required function. If both valves failed to close, a loss of reactor coolant in the form of water discharged from the CRD system would occur. The amount of water discharged will be relatively small, and is more of a concern from the standpoint of contamination to the Secondary Containment rather than a loss of reactor water inventory. A structural failure of the SDV, which
bounds this case of an open SDV vent or drain line, has been previously
evaluated in NUREG-0808, ``Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping.'' In this evaluation, the NRC concluded that, for a bounding leakage case corresponding to a rupture of the SDV, the offsite doses would be well within the limits of 10CF100, and that adequate core cooling would be maintained.

Deletion of the requirement that the SDV be determined OPERABLE by testing the SDV vent and drain valves, as proposed in this TS Change Request, will have an insignificant effect on the probability of occurrence of malfunction of any plant equipment. The conditions in the
SDV at the time of vent and drain valve closure are not appreciably different whether a scram is initiated from power operation or during shutdown conditions. In addition, this proposed TS change eliminates the potential need for an additional startup and shutdown cycle, along with the associated challenges to all systems and components, that would be required to satisfy the current TS requirements in the event a
unit were to trip off-line shortly before a planned outage when the surveillance was scheduled to be performed. Furthermore, this proposed TS changes does not affect the testing frequency for the valves.

This proposed TS change will not result in appreciably different conditions experienced by the valves as they close, and their ability to re-open is confirmed following each reactor scram from power conditions. The consequences resulting from a failed closed or failed open SDV vent or drain line have been evaluated and determined not to result in offsite doses that would exceed the limits specified in 10CFR100, or jeopardize adequate reactor core cooling capability. Therefore, the consequences of a malfunction of equipment important to

safety previously evaluated is not increased.
Therefore, the proposed TS change does not involve an increase in the probability or consequences of an accident previously evaluated.
2. The proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
The SDV is not an accident initiator. Deletion of the requirement that the SDV be determined OPERABLE by testing the SDV vent and drain valves from a configuration of less than or equal to 50% rod density, as proposed, will not create the possibility of a different type [of] accident than any previously evaluated.
No plant equipment is added or deleted as a result of this proposed change. Since the initial conditions of pressure, temperature, and CRD system discharge flowrate have **no** appreciable effect on the SDV vent and drain valve performance, **no** different type of malfunction of any equipment important to safety is created.
Therefore, the proposed TS change does not create the possibility of a new different kind of accident from any previously evaluated.
3. The proposed TS change does not involve a significant reduction in a margin of safety.
Since the initial test conditions of pressure, temperature, and CRD discharge flowrate will have **no** appreciable effect on the SDV vent and drain valve performance, conducting the surveillance test during shutdown conditions, as specified in this proposed TS change, will not affect the validity of the surveillance results with respect to the operability of the SDV to perform its intended safety function. Furthermore, every reactor scram is a serious plant transient and a potential challenge to safety-related systems and equipment. The potential decrease in future scrams which could result from this proposed TS change will represent an improvement in overall safety.
Therefore, the proposed TS change does not involve a reduction in a margin of safety.
Section 4: Optional Method of Scram Timing
1. The proposed Technical Specification (TS) changes involves a significant increase in the probability or consequences of an accident previously evaluated.
Scram testing control rods at zero reactor coolant pressure will not increase the probability of any control rod related transient or accident discussed in the UFSAR [Updated Final Safety Analysis Report].
UFSAR Sections 15.4.1.1 and 15.4.1.2 discuss the consequences of

inadvertent reactivity insertion errors due to the withdrawal of one or
more control rods. The probability of one of these events occurring is a function of operator error and equipment malfunction and is not related to scram insertion times.

An inadvertent reactivity insertion error is prevented by existing system hardware interlocks and procedural controls that are not affected by scram time testing, e.g., core design, control and design, one-rod-out interlocks, refueling interlocks, control rod sequence designations, and neutron monitoring systems.

USFAR Section 15.4.9 discusses the control rod drop accident (CRDA). The CRDA assumes that a control rod suddenly drops out of the core due to equipment malfunction. The probability of occurrence of this accident is based on an equipment malfunction and is not affected by scram testing.

Engineering analysis and control rod scram test data demonstrate that a control rod drive that will meet the 2.0 second, scram insertion
time, test criteria at zero reactor coolant pressure will also meet all
scram insertion criteria during reactor startup and up to 40% rated thermal power.

The 2.0 second criterion was chosen to conservatively envelop scram
time criteria and reactivity insertion criteria during reactor startup and up to 40% rated power conditions. Therefore, scram testing affected
control rods at zero reactor pressure will not increase the consequences of an accident previously evaluated.

UFSAR Sections 15.4.1.1 and 15.4.1.2 evaluate reactivity insertion transients at low power conditions due to inadvertent control rod withdrawal errors. The UFSAR concludes that rod withdrawal errors at low power are adequately precluded by refueling interlocks, rod worth minimizer, operating procedures, core design, and control rod hardware design. However, should operator errors followed by equipment malfunctions result in an inadvertent criticality event, the IRMs would
provide the necessary rod blocks or reactor scram to preclude the operational transient. Scram insertion time limits for the continuous rod withdrawal error during startup is 5.0 seconds. This scram time criterion will be met by a control rod that scrams within 2.0 seconds at zero reactor pressure. The 2.0 second scram criterion was established to ensure that affected control rods will meet scram

requirements from zero reactor pressure up to 40% core thermal power.
Also, during low power operation (UFSAR Subsection 15.4.1.2) the rod worth minimizer (RWM) prevents the operator from selecting and withdrawing an out-of-sequence control rod. During reactor operation in the power range (UFSAR subsection 15.4.2) the rod block monitor (RBM) prevents a rod withdrawal error by inhibiting inadvertent control rod withdrawal. The RWM and RBM do not rely on a scram function to mitigation the consequences of a rod withdrawal error, and therefore the consequences of an accident evaluated in the UFSAR will not be affected by the proposed changes to the Technical Specifications.

The consequences of a control rod drop accident (UFSAR Section 15.4.9) would not be affected by scram testing a control rod at zero reactor pressure. The design basis accident of the rod drop accident assumes that control rods scram within 5.0 seconds. This 5.0 second scram test requirement will be met by control rods that meet the 2.0 second criterion at zero reactor pressure.

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

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

The changes to the Technical Specifications will allow control rods to be scram tested at zero reactor pressure and then again at rated reactor pressure prior to achieving 40% rated reactor power. **No** new types of accidents will be introduced since control rods that meet the 2.0 second scram criterion at zero reactor pressure will also meet all scram test criteria during reactor startup and at rated reactor pressure.

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

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

The basis for shutdown margin (TS Bases 3/4.1.1) states that the reactor shall be made subcritical by all certain margin in all operating and shutdown conditions. The proposed changes to the Technical Specifications will not affect the shutdown margin requirements. Adequate shutdown margin is assured by core design, the one-rod-out interlock, and administrative controls.

The basis for the control rod insertion times (TS Bases 3/4.1.3) states that the scram times are to be consistent with those used in

the
transient and accident analysis. The proposed Technical Specifications changes will add an additional scram test verification for affected control rods at zero reactor pressure. The zero reactor pressure scram limit (2.0 seconds) was designed to ensure that the scram times assumed
in the transient analysis will remain bounding from zero reactor pressure up to 40% rated core thermal power.

The basis for the control rod drop accident (TS Bases 3/4.1.3) states that the potential effects of a CRDA are limited. The proposed Technical Specifications changes will not effect the control rod drop results as the changes do not affect the reactivity of the rod or the rod drop velocity. The CRDA analysis is based on a 5.0 second scram insertion time criterion. The 2.0 second time criterion was established
to ensure that the 5.0 second scram time criterion was valid from zero reactor pressure to 950 psig reactor pressure.

The basis for MCPR limits (TS Bases 3/4.1.3 and 2.3) states the CRD
system must bring the reactor subsubcritical at a rate fast enough to prevent MCPR from becoming less than the fuel cladding safety limit during the limiting power transient analyzed in the UFSAR. The proposed
changes to the Technical Specifications will not affect the scram insertion rates that are used as input to the transient analysis. The zero reactor pressure scram limit of 2.0 seconds was developed to ensure that the control rods would meet their design scram insertion times from zero reactor pressure up to 40% rated power.

The proposed changes to the Technical Specifications will not increase the probability of inadvertent criticality because the changes
do not affect the reactivity worth of control rods.

Therefore, the proposed TS changes do not involve a reduction in a margin of safety.

Section 5: Definition of Core Alteration

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

The proposed definition change removes the requirement to have a SRO or LSRO supervise control rod withdrawal in an off-loaded cell (i.e. **no** fuel assemblies). The evaluated accident potentially affected by this change is a control rod movement error during refueling resulting in inadvertent criticality. The supervision by a SRO or LSRO does not solely preclude inadvertent criticality and was not relied

upon in the accident analysis contained in Section 15.4 of the LGS Updated Final Safety Analysis Report (UFSAR). The LGS reactor core is designed to have adequate shutdown margin with the highest-reactivity-worth control rod withdrawn. The withdrawal of a second rod with fuel assemblies loaded in the associated cell is prevented by a combination of the refueling, one-rod-out interlock, and the Limiting Conditions for Operation (LCO) requirement of TS 3.9.10.2. The LCO requirements ensure adequate shutdown margin is present prior to control rod withdrawal. This is accomplished by testing during startup following a refueling outage or by analytical calculations during refueling. The refueling interlock will provide a rod block upon an attempt to withdraw a second control rod and is required to be operable in accordance with TS 3.9.10.2 except for rods which have **no** fuel assemblies in the associated cell. The removal of the fuel assemblies from a cell eliminates the need for the reactivity control function of the associated rod. The physical removal of a control blade from the core by means of the refueling floor, first requires the removal of the
four associated fuel assemblies in the cell. This design inherently prevents inadvertent criticality. Finally, this change is consistent with NUREG-1433 ``Standard Technical Specifications.'' Since current analysis permits the withdrawal of a control rod blade, provided the associated cell is unloaded, and refueling mode interlocks, administrative TS requirements and the physical design of the control blade and fuel cell, which preclude inadvertent criticality, will remain unchanged, this proposed change to the TS definition of CORE ALTERATION will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The LGS UFSAR currently permits control rod withdrawal and or removal, provided there are **no** fuel assemblies in the associated fuel cell. The definition change removes the requirement to have a SRO or LSRO supervise rod withdrawal in an off-loaded cell. The change potentially [a]ffects a control rod movement error during refueling resulting in inadvertent criticality which has been previously evaluated. In addition, the proposed change will make **no** physical changes to equipment or remove administrative controls which solely preclude inadvertent criticality. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

    The LGS TS bases address reactivity concerns, radiological
releases, control rods, and monitoring of the facility related to this
change. With the four fuel assemblies removed from a cell, the control
rod/blade in the associated cell has **no** reactivity function. The
reactivity issues addressed by TS are therefore unaffected. The rod/
blade coupling integrity is maintained by the requirement to perform a
coupling check following maintenance. Section 15.4 of the UFSAR states
that there are **no** radiological releases in association with a rod
withdrawal error during refueling. This conclusion is maintained by
the
administrative requirements of TS 3.9.10.2, the refueling interlocks
for one-rod-out, and the physical design of the blade and cell.
Lastly,
the TS requirements for Emergency Core Cooling, Plant System,
Containment, and Electrical Power Distribution System, which provide
the systems necessary to mitigate the effects of a radiological
release
during control rod movement in an unloaded cell were reviewed and were
found not to be adversely [a]ffected by the proposed change.
Therefore,
this change will not involve a significant reduction in a margin of
safety.

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

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

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    NRC Project Director: John F. Stolz.

Philadelphia Electric Company

Docket Nos. 50-352 and 50-353

    Limerick Generating Station, Units 1 and 2, Montgomery County,
Pennsylvania.

    Date of amendment request: August 31, 1994.

    Description of amendment request: The proposed amendments, which
are consistent with the Improved Standard Technical Specifications

(NUREG-1433), involve the following six (6) sections of TS changes:

Section 1: Relocation of Turbine Overspeed Protection System Requirements;
Section 2: Relocation of Primary Containment Conductor Protection Devices Requirements;
Section 3: Feedwater/Main Turbine Trip System Actuation Instrumentation Requirements;
Section 4: Permit Operability of Low Pressure Coolant Injection While Aligned to Shutdown Cooling;
Section 5: Remove Temperature Requirement for Operational Condition [OPCON] 5; and
Section 6: Reduce Frequency of Alternate Decay Heat Demonstration

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:

Section 1: Relocation of Turbine Overspeed Protection System Requirements

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

The proposed change relocates requirements from the TS, to licensee controlled documents. The licensee controlled documents containing the relocated requirements will be maintained using the provisions of 10 CFR 50.59 and are subject to the change control process in the Administrative Controls Section 6.0 of the TS. Since changes to licensee controlled documents will be evaluated per 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 will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This change relocates requirements to licensee controlled documents. This change will not alter the plant configuration (**no** new or different type of equipment will be installed) or make changes in methods governing normal plant operation. This change will not impose different requirements and adequate control of information will be maintained. This change will not alter assumptions made in the safety analysis and licensing basis. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change relocates requirements from the TS to licensee controlled documents. This change will not reduce a margin of safety since it has **no** impact on any safety analysis assumptions. In addition,
the requirements to be transferred from the TS to licensee controlled documents are the same as the existing Technical Specifications. Since any future changes to these licensee controlled documents will be evaluated per the requirements of 10 CFR 50.59, **no** reduction (significant or insignificant) in [a] margin of safety will be allowed.
Therefore, this change will not involve a significant reduction in a margin of safety.

The existing requirements for NRC review and approval of revisions,
in accordance with 10 CFR 50.59, to these details and requirements proposed for relocation, does not have a specific margin of safety upon
which to evaluate. However, since the proposed change is inconsistent with the BWR [boiling-water reactor] Improved Standard Technical Specifications (NUREG-1433 approved by the NRC Staff) and the change controls for proposed relocated details and requirements provide an equivalent level of regulatory authority, revising the TS to reflect the approved level of detail and requirements ensures **no** reduction to the margin of safety.

Section 2: Relocation of Primary Containment Conductor Protection Devices Requirements

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

This proposed change relocates requirements from the TS to licensee
controlled documents. The licensee controlled documents containing the relocated requirements will be maintained using the provisions of 10 CFR 50.59 and are subject to the change control process in the Administrative Controls Section 6.0 of the TS. Since changes to these licensee controlled documents will be evaluated per 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 will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This change relocates requirements to licensee controlled documents. This change will not alter the plant configuration (**no** new or different type of equipment will be installed) or make changes in methods governing plant operation. This change will not impose different requirements and adequate control of information will be maintained. This change will not alter assumptions made in the safety analysis and licensing basis. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change relocates requirements from the TS to licensee controlled documents. This change will not reduce a margin of safety since it has **no** impact on any safety analysis assumptions. In addition,
the requirements to be transferred from the TS to the licensee controlled documents are the same as the existing TS. Since any future changes to these licensee controlled documents will be evaluated per the requirements of 10 CFR 50.59, **no** reduction (significant or insignificant) in [a] margin of safety will be allowed. Therefore, this
change will not involve a significant reduction in a margin of safety.

The existing requirements for NRC review and approval of revisions,
in accordance with 10 CFR 50.59, to these details and requirements proposed for relocation, does not have a specific margin of safety upon
which to evaluate. However, since the proposed change is consistent with the BWR Improved Standard TS (NUREG-1433 approved by the NRC Staff) and the change controls for proposed relocated details and requirements provide an equivalent level of regulatory authority, revising the TS to reflect the approved level of detail and requirements ensures **no** reduction to the margin of safety.

Section 3: Feedwater/Main Turbine Trip System Actuation Instrumentation
Requirements

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

For the proposed TS change, in the event of a Reactor Vessel Water Level--High Level 8 transient, operator action per existing plant procedures would terminate the event and prevent damage to the Main/

RFP
[reactor feed pump] Turbine due to water carry over. The Main/RFP
Turbine do not serve a safety function, also at <25% [Rated Thermal
Power] RTP a level 8 transient event will not cause a reactor scram.
An
analysis of information in the bases for APLHGR [average planar linear
heat generation rate] and MCPR [minimum critical power ratio] has
shown
that a sufficient margin to core safety limit exist, so fuel integrity
levels are not violated. Therefore, the proposed TS change does not
involve an increase in the probability or consequences of an accident
previously evaluated.

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

Should the feedwater/main turbine trip system, Reactor Vessel
Water
Level-High Level 8, not actuate in OPCON 1 at <25% RTP, operator
action
per existing plant startup procedures would protect the Main/RFP
turbines. If operator action is not performed, damage to Balance of
Plant, non-safety related equipment could occur. High Reactor Vessel
Water Level is not a concern for reactor core safety at <25% RTP.
Therefore, the proposed TS change does not create the possibility of a
new or different kind of accident from any accident previously
evaluated.

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

The proposed TS change, which revises the feedwater/main turbine trip system actuation instrumentation, Reactor Vessel Water Level-High Level 8, operability requirements, does not affect the TS bases. The trips are designed to protect Balance of Plant Equipment at all Rate Power Levels. The Reactor Vessel Water Level-High Level 8 trips also protects fuel integrity at >25% RTP. Therefore, the proposed TS change to the operability requirements for the feedwater/main turbine trip system actuation instrumentation does not involve a reduction in a margin of safety.

Section 4: Permit Operability of Low Pressure Coolant Injection While Aligned to Shutdown Cooling

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

The LPCI [low pressure coolant injection] mode of RHR is an accident mitigator, not an initiator. Currently, the LPCI mode of RHR

is an automatic Emergency Core Cooling System (ECCS) function during OPCONs 4 and 5. However, shutdown cooling has been an accident initiator in many industry events. Reliance on this loop of RHR for LPCI does not increase the probability of an accident in shutdown cooling, but the alignment for LPCI will, in itself, terminate the draindown event by exiting the shutdown cooling mode. This proposed change will permit the operability of one LPCI subsystem while the components of that subsystem are aligned and operating in the Shutdown Cooling mode of RHR, provided all other components of that subsystem are operable and can be manually realigned from the Main Control Room, if required. The required number of operable Emergency Core Cooling Systems (ECCS) remains unchanged, thus maintaining the TS required subsystem redundancy (TS Section 3.5.2 requires two operable ECCS subsystems with exception for Reactor level). With this change, the required number of LPCI subsystems are capable of performing their function of limiting and/or mitigating the consequences of an accident,
by allowing the manual alignment of one LPCI subsystem, during OPCONs 4
and 5. This allowance is justified since the change only applies to OPCONs 4 and 5, when reactor temperature, and associated heat loads are
sufficiently low to provide the operator sufficient time to perform the
manual realignment, from the Main Control Room, of the RHR pump suction
valves and restart of the pump following LPCI injection conditions. Similar allowances for LPCI are currently permitted during OPCON 3, since the decay heat loads are significantly reduced compared to OPCON 1, which is the mode of operation under which ECCS capability is analyzed (Section 6.3 of the LGS [Limerick Generating Station] Updated Final Safety Analysis Report (UFSAR)). The change will not increase the
probability of occurrence or consequences of a malfunction of equipment
since there will be **no** physical changes made to plant equipment nor the
method of their operation that would result in an unanalyzed condition.
PECO Energy [Philadelphia Electric Company] evaluated the need for manual realignment of the pump minimum flow path since operating in Shutdown Cooling typically results in the isolation of the pump minimum

flow path to prevent inadvertent draining of the reactor vessel. The associated pump is still operable since this change is limited to OPCONs 4 and 5, when reactor pressure is sufficiently low to allow immediate injection to the reactor vessel without a minimum flow path. In situations, while in OPCON 4, where reactor pressure may not be sufficiently low to allow injection, the RHR system will not be aligned for Shutdown Cooling, since the reactor vessel pressure will be greater than the RHR ``cut-in'' permissive pressure. In addition, Administrative Controls are currently in place to realign RHR to the LPCI mode for planned pressure increases. Finally, this change is consistent with NUREG-1433 ``Standard Technical Specifications.'' Therefore, these changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The LPCI mode of RHR is an accident mitigator, not an initiator. This change will not reduce the number of required ECCS during OPCONs 4 and 5. This change will permit the operability of one LPCI subsystem while the components of that subsystem are aligned and operating in the Shutdown Cooling mode of RHR. The change does not alter current methods of plant operation nor does the change make a physical change to plant equipment resulting in an unanalyzed malfunction of equipment. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The basis of TS Section 3.5.2 is to ensure sufficient ECCS capacity to maintain core cooling in OPCONs 4 and 5. This proposed change does not affect the required number of ECCS during OPCONs 4 and 5; therefore, adequate capability through subsystem redundancy is maintained. The amount of time required to obtain rated LPCI conditions is increased due to the manual realignment, from the Main Control Room, of the suction valves and restart of the RHR pump following LPCI injection conditions. This change is in conformance with the current

TS
bases, since the operator has sufficient time to perform the manual realignment, during OPCONs 4 and 5, ensuring sufficient ECCS capability
to maintain core coverage. In addition, NUREG-1433 BASES states, in part, ``One LPCI subsystem may be aligned for decay heat removal and considered OPERABLE for the ECCS function, if it can be manually realigned (remote or local) to the LPCI mode and is not otherwise inoperable. Because of low pressure and low temperature conditions in MODES 4 and 5, sufficient time will be available to manually align and initiate LPCI subsystem operation to provide core cooling prior to postulated fuel uncover.'' Therefore, this change will not involve a significant reduction in a margin of safety.

Section 5: Remove Temperature Requirement for Operational Condition 5

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

The proposed TS change does not involve a physical change in the configuration of any systems important to safety. The elimination of a temperature requirement from the definition of OPCON 5 was reviewed for
potential effect on reactor coolant system materials and for potential effect on reactivity. This TS change does not result in system temperature and pressure change or reactivity changes not previously analyzed. The reactor pressure vessel will still be restricted to the temperature and pressure limits of TS Section 3/4.4.6 which includes heatup/cooldown rates and minimum boltup limits. The reactor pressure vessel temperature and pressure limits will still ensure proper protection of the reactor coolant system materials. Therefore, this TS change does not increase the probability or consequences of an accident
previously evaluated.

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

The proposed TS change does not involve any physical change in plant configuration, and reactor coolant system temperature and pressure are still restricted per TS Selection 3/4.4.6. The decrease in
moderator density corresponding to the potential change in temperature (i.e., above 140 deg.F and below 200 deg.F) would have a negligible, however conservative effect on shutdown margin. Therefore, this TS change does not create the possibility of a new or different kind of

accident from any accident previously evaluated.

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

This proposed TS change does not change the reactor coolant system material restrictions as defined in TS Section 3/4.4.6. Therefore, the reactor pressure vessel will still be maintained under the current temperature and pressure restrictions as well as the current boltup limits.

The decrease in moderator density corresponding to the potential temperature change from 140 deg.F to 200 deg.F is insignificant and would afford approximately the same moderator effect. Therefore, shutdown margin could only be improved (although marginally) at these evaluated temperatures. The actual coolant temperature will be administratively controlled to provide for personnel safety. Therefore,
this change will not involve a reduction in a margin of safety.

Section 6: Reduce Frequency of Alternate Decay Heat Demonstration

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

The proposed TS change does not involve any physical changes to plant systems or equipment. This proposed TS change will allow the use of either an ``analytical approach'' (i.e., calculation) or ``demonstration'' to ensure the operability of an alternate decay heat removal method. This proposed TS change does not involve any physical changes to plant systems or components, nor does it affect the capability, availability, or operability of any decay heat removal systems/methods (e.g., Shutdown Cooling). The Shutdown Cooling mode of operation of the Residual Heat Removal (RHR) system, and Residual Heat Removal Service Water (RHRSW) system, are not impacted by this proposed
TS change, and will continue to function as designed to remove decay heat loads from the reactor primary coolant system. The RHRSW system and various modes of operation of the RHR system, e.g., Low Pressure Coolant Injection (LPCI) are not accident initiators, since these systems function to mitigate the consequences of an accident. This proposed TS change is consistent with the criteria delineated in the Improved Standard TS (i.e., NUREG-1433, ``Standard Technical Specifications, General Electric Plants, BWR/4,'' dated September 28, 1992).

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

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

This proposed TS change does not involve any physical changes to plant systems or equipment. The proposed TS change will allow the use of a ``calculation'' or ``demonstration'' as the means for determining the operability of an alternate decay heat removal method. The proposed TS change does not involve any physical changes to plant systems or equipment. This proposed TS change will not affect the operation of the Shutdown Cooling mode of the RHR system. This mode of operation will continue to function as designed to remove decay heat loads from the reactor primary coolant system. This proposed TS change will not impact the operation of the other modes of operation of the RHR system (e.g., LPCI), nor will it affect the operation of the RHRSW system. These systems will continue to function as designed, which is to mitigate the consequences of an accident. This proposed TS change will not introduce the potential for equipment malfunctions or failures. This proposed TS change is consistent with the criteria delineated in the Improved Standard TS (i.e., NUREG-1433).

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

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

The proposed change to the TS does not involve any physical changes to plant systems or equipment. This proposed TS change does not make any physical modifications to plant systems or equipment, and is consistent with the criteria delineated in the Improved Standard TS (i.e., NUREG-1433). The proposed TS change will not impact any mode of operation of the RHR system or the RHRSW system.

This proposed TS change involves revising TS ACTION statements, and associated supporting Bases sections, to allow for the use of a ``calculation'' or ``demonstration'' to ensure the operability of an alternate decay heat removal method. The bases for the TS sections affected by this proposed change indicate that sufficient heat removal capability, system redundancy, and coolant circulation will be available to facilitate decay heat removal and mixing to assure accurate temperature indication.

This proposed TS change does not affect the function or availability of any decay heat removal system or method.

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

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

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

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: John F. Stolz.

Power Authority of the State of New York

Docket **No**. 50-333

James A. FitzPatrick Nuclear Power Plant, Oswego County, New York.

Date of amendment request: October 3, 1994.

Description of amendment request: The proposed amendment would extend the functional test intervals and allowable out-of-service times
for some of the instruments subject to requirements of the Technical Specifications (TSs). These proposed changes are based upon NRC-approved Licensing Topical Reports prepared under the direction of the Boiling Water Reactors Owners Group and intended to enhance plant safety by reducing the potential for test related scrams, excessive test cycles on equipment, and operator errors. The proposed amendment would also: (1) Remove the Average Power Range Monitor (APRM) downscale
scram function from the TSs, remove instrument response time values from the TSs in accordance with Generic Letter 93-08, and incorporate various editorial changes and clarifications into the TSs. The proposed
amendment involves reactor protection system, primary containment isolation, emergency core cooling, control rod block, and anticipated transient without scram recirculation pump trip 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:

Operation of the FitzPatrick plant in accordance with the proposed Amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated because:

a. Incorporate STI [Surveillance Test Interval] and AOT [Allowable Out-Of-Service Time] Improvement--Category 1

The proposed changes are limited to an extension of the surveillance testing intervals and allowable out-of-service times of plant instrumentation. The changes do not introduce any new modes of plant operation, make any physical changes, or alter any operational setpoints. Therefore, the changes do not degrade the performance of any safety system assumed to function in the accident analysis. Consequently, there is **no** effect on the probability of occurrence of an accident.

Regarding the consequences of an accident, the GE [General Electric Company] Licensing Topical Reports (References 1 through 7) concluded that the proposed extensions in the STI and AOT for the safety system instrumentation results in an insignificant change in the core damage frequency. The extension of the STI/AOTs results in a slight increase in the unavailability of the safety functions. However, this effect is offset by a reduction in the probability of inadvertent plant trips due to reduced testing. The overall effect on the probability of an accident is negligible. While the effects of reducing unnecessary cycles on safety system instrumentation is not quantifiable, the effect will be to further reduce the core damage frequency. The NRC concurred in their SERs [Safety Evaluation Reports] (References 8 through 15) with these conclusions. Consequently, there is not a significant increase in the consequences of an accident.

b. Relocation of the Instrument Response Time Limits--Category 2

The change involves the use of an alternate regulatory process for controlling the instrument response time limits. The change does not introduce any new modes of plant operation, make any physical changes, alter any operational setpoints, or change the surveillance requirements.

c. Delete APRM Downscale Scram--Category 3

The design basis accident applicable to the startup power region is
the Control Rod Drop Accident (CRDA). The FSAR [Final Safety Analysis Report] does not credit the APRM downscale scram in the termination of this accident. Accident mitigation is provided by the APRM fixed high neutron flux scram. Therefore, elimination of this scram functions has **no** adverse affect on previously evaluated accidents.

d. Miscellaneous Changes--Category 4

The changes do not introduce any new modes of plant operation, make
any physical changes, or alter any operational setpoints. The changes involve enhancements that clarify the Technical Specification requirements.

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

a. Incorporate STI and AOT Improvements--Category 1

The proposed changes do not introduce any new accident initiators or failure mechanisms since the changes do not introduce any new modes of plant operation, make any physical changes, or alter any operational
setpoints. The changes reduce the probability of accidents initiated by
test-induced plant trips.

b. Relocation of the Response Time Limits--Category 2

The change involves the use of an alternate process for controlling
the instrument response time limits. The change does not introduce any accident initiators since it does not involve any new modes of plant operation, make any physical changes, alter any operational setpoints, or change the surveillance requirements.

c. Delete APRM Downscale Scram--Category 3

Scram functions are intended to shutdown the reactor following transients or accidents and their removal does not introduce an accident initiator. The limiting accident evaluated in the FSAR for the startup power region is the control rod drop accident. This accident is assumed to occur irrespective of the scram functions provided to terminate this accident.

d. Miscellaneous Changes--Category 4

The changes do not introduce any new accident initiators or failure mechanisms since the changes do not alter the physical characteristics of any plant system or component. The changes involve enhancements that clarify the Technical Specification requirements.

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

a. Incorporate STI and AOT Improvements--Category 1

The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The affected instrumentation setpoints already account for the effects of drift and include sufficient allowance for an extension in the STIs. The evaluations presented in the referenced Licensing Topical Reports concluded that the overall effect of the proposed changes provides a net increase in plant safety.
The improvement is achieved by reducing the potential for: (a) Test related plant scrams (reduced challenges to plant shutdown systems and improved plant availability); (b) excessive test cycles on equipment (reduced wear-out potential); (c) operator errors (AOT provides reasonable time for making repairs and tests); (d) scrams that occur when inoperable channels are tripped because insufficient repair time exists; and (e) diversion of plant personnel and resources on unnecessary testing (potential safety and operational improvement).

b. Relocation of the Response Time Limits--Category 2

The change involves the use of an alternate regulatory process for controlling the instrument response time limits. The change does not introduce any new modes of plant operation, make any physical changes, alter any operational setpoints, or change the surveillance requirements.

c. Delete APRM Downscale Scram--Category 3

The only scram function that the UFSAR [Updated Final Safety Analysis Report] takes credit for in the mitigation of the limiting accident (control rod drop accident) is the APRM 15% power fixed high neutron flux scram. This scram function, as well as the IRM [Intermediate Range Monitor] high flux scram function in the startup mode which could also terminate this accident, are not affected by this change. Only the APRM downscale scram, for which the UFSAR takes **no** credit in the termination of any analyzed event, is eliminated by this change. The APRM downscale control rod block is not affected by this change. Elimination of the APRM downscale scram will avoid the need to operate the plant in a ``half scram'' condition for certain IRM/APRM channel bypass combinations, therefore eliminating the potential for an inadvertent plant transient. For these reasons, the change does not involve a significant reduction in the safety margin.

d. Miscellaneous Changes--Category 4

The changes assure compliance with the Technical Specifications by improving its clarity and accuracy. For these reasons the changes will improve the plant's margin of safety.

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

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

Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, New York 10019.

NRC Project Director: Ledyard B. Marsh.

Power Authority of the State of New York

Docket **No**. 50-333

James A. FitzPatrick Nuclear Power Plant, Oswego County, New York.

Date of amendment request: October 7, 1994.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 4.6.E.4 to establish that the manual cycling of reactor coolant system (RCS) safety/relief valves (SRVs) during plant startups is to be accomplished within 12 hours after steam pressure and flow are adequate to perform the testing. TS 4.6.E.4 currently requires this testing to be performed within 12 hours of continuous power operation at a reactor steam dome pressure of at least 940 psig. This change was proposed to minimize the potential for undesirable pressure transients in the RCS. The amendment would also make several editorial changes to clarify the intent of TS's involving SRV valve testing and performance requirements.

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

Operation of the James A. FitzPatrick Nuclear Power Plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed changes do not change the test method or conditions under which valve testing may be performed and there is **no** affect on assumptions used for previously analyzed accidents. The original operating license for FitzPatrick did not specify any time limit for completing manual testing of the safety/relief valves.

2. Create the possibility of a new or different kind of accident from those previously evaluated because the proposed amendment does not involve any modification of plant equipment or changes in plant operating conditions.

3. Involve a significant reduction in the margin of safety because the proposed amendment makes **no** changes to the operability of performance requirements for the safety/relief valves including the ADS [Automatic Depressurization System] function. Valve lift setpoints and the minimum number of operable valves required are not affected.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the

amendment request involves **no** significant hazards consideration.
Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.
Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, New York 10019.
NRC Project Director: Ledyard B. Marsh.

Public Service Electric & Gas Company

Docket Nos. 50-272 and 50-311

Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey.
Date of amendment request: September 9, 1994.
Description of amendment request: The proposed amendment modifies the visual inspection for snubbers in the Technical Specifications and is consistent with the guidance provided in Generic Letter 90-09.
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 involve **no** hardware changes, **no** changes to the operation of snubbers, and does not change the ability of the snubbers to perform their intended functions. Visual inspection of snubbers is a separate process that complements the functional testing program. The NRC has concluded that functional testing of snubbers provides a 95 percent confidence level and 90 to 100 percent of the snubbers will operate within the specified acceptance limits. Any change in the visual inspection frequency will not have any significant impact on the operability of the snubbers.
2. Will not create the possibility of a new or different kind of accident from any previously evaluated.
The proposed changes will not result in an unanalyzed condition. Replacing the current method of determining visual surveillance intervals with a new method approved by the NRC in Generic Letter 90-09

will not change the level of confidence in snubber operability. A new procedure for determining visual inspection frequencies will not result
in an unreviewed failure mechanism.

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

The proposed changes incorporate the alternate Technical Specification requirements for visual inspection of snubbers identified
in Generic Letter 90-09. The alternate visual inspection criteria consider the size of the category of snubbers when evaluating inspection intervals due to failure rates. Since the functional testing
requirements remain unchanged and do not reduce the operability confidence levels, there is **no** resultant change in any margins of safety.

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

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

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

NRC Project Director: John F. Stolz.

Public Service Electric & Gas Company

Docket Nos. 50-272 and 50-311

Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey.

Date of amendment request: September 20, 1994

Description of amendment request: The proposed amendment modifies the Technical Specifications for auxiliary feedwater to reduce the secondary side steam pressure required for testing the steam turbine driven auxiliary feedwater pump (AFW). The proposed amendment also clarifies the time required to perform the steam turbine driven auxiliary feedwater pump surveillance test when entering Mode 3.

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

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

The proposed change to the minimum required test pressure for the steam turbine driven AFW pump does not affect the operation of the pump during conditions when it is required to performed its safety function.

The clarification that the secondary side steam pressure is steam generator pressure is editorial. Reduced Tavg and increased steam generator tube plugging will affect the normal operating secondary side steam pressure.

However, the zero load secondary side steam pressure is not affected, therefore, the conditions in which the steam turbine driven AFW pump will be required to perform its safety function are not changed.

Providing a specific time frame in which to perform the surveillance test after attaining the required steam pressure ensures that the test will be performed in a timely manner. The time frame specified is consistent with NUREG-1431, Standard Technical Specifications--Westinghouse Plants.

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

2. Create the possibility of a new or different kind of accident.

The proposed changes do not change system configurations, plant equipment, or analysis. Therefore, the proposed changes will not increase the possibility of a new or different kind of accident from any accident previously identified.

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

The proposed change to the minimum required steam pressure will not affect the heat removal capability of the AFW System. Therefore, the value assumed in the safety analysis is not changed. The change to the specification 4.0.4 exemption to provide a specific time period does not affect any margins of safety. Therefore, these changes do not involve a significant reduction in any margin of safety.

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

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

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

NRC Project Director: John F. Stolz.

Public Service Electric & Gas Company

Docket Nos. 50-272 and 50-311

Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey.

Date of amendment request: September 20, 1994.

Description of amendment request: These proposed changes would adopt the Westinghouse Standard Technical Specifications (NUREG-1431) Channel Functional Test surveillance frequency for the Manual Reactor Trip Switches and for the Reactor Trip Breakers (RTB) and relocate RTB maintenance requirements from the Technical Specifications to the Salem, Units 1 and 2, Updated Final 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:

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

The proposed changes do not affect accident conditions or assumptions. They change the existing surveillance test and their frequencies to make them consistent with industry standards, and relocate maintenance requirements to the UFSAR [Updated Final Safety Analysis Report].

The changes, for the Manual Reactor Trip Switch and Reactor Trip Breaker (RTB) CHANNEL FUNCTIONAL TEST frequency, incorporate the established Westinghouse STS surveillance frequencies. These surveillance frequencies have received previous NRC review and generic approval via the issuance of NUREG-1431. The Westinghouse STS does not require Channel Functional Test for the Manual Reactor Trip Switches or
the RTB prior to each reactor startup.

The addition of the RTB shunt trip feature for automatic reactor trips, the improved RTB maintenance activities developed over the past several years, and the implementation of 10 CFR 50.62 requirements have

improved RTB reliability. These features are unaffected by the proposed
changes. Excessive RTB testing results in increased component wear and possibly reduced component life. Testing the RTBs with associated logic
trains reduces the potential for human errors and associated plant transients.

The consequences of accidents previously evaluated are unaffected by the proposed changes.

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

The proposed changes do not modify any system or equipment, nor alter any process function. The Manual Reactor Trip Switch and RTB functionality remains unchanged. Therefore these changes do not create a new or un-evaluated accident or operating condition.

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

The proposed changes adopt the NRC approved Westinghouse STS surveillance testing frequencies to maintain RTB reliability. Reduced testing at power, consistent with the associated logic train test frequency, reduces the potential for inadvertent actuation and personnel errors. Thus, the proposed changes enhance plant safety.

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

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

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

NRC Project Director: John F. Stolz.

Saxton Nuclear Experimental Corporation

Docket **No**. 50-146

Saxton Nuclear Facility, Bedford County, Pennsylvania.

Date of amendment request: August 8, 1994. This supersedes the request dated June 23, 1993.

Description of amendment request: The proposed amendment would revise the technical specifications to allow characterization activities related to the decommissioning of the Saxton Nuclear

Facility and add administrative activities associated with the characterization activities.

Basis for Proposed **No** Significant Hazards Consideration Determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of **no** significant hazards consideration, which is presented below:

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

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

The activities associated with characterization of the facility will have a minimum impact on the physical condition of the containment vessel as it relates to the risk of fire and has **no** effect on the risk of flooding.

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

In its present condition, the only accidents applicable to the site are fire, flood, and radiological hazard. The possibility of a new or different type of accident than that previously evaluated in the FSAR will not be created by the implementation of activities permitted by the approval of this amendment request.

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

No margins of safety relevant to the equipment at the facility exist. Activities involved in characterization will not involve a reduction in a margin of safety.

The NRC staff has reviewed the analysis of the licensee 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: Saxton Community Library, 911 Church Street, Saxton, Pennsylvania 16678.

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

NRC Project Director: Seymour H. Weiss.

Southern California Edison Company, et al.

[Docket Nos. 50-361 and 50-362

San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San

Diego
County, California.
    Date of amendment request: August 26, 1994.
    Description of amendment requests: The licensee proposes to revise Technical Specification (TS) 3/4.7.5, ``Control Room Emergency Air Cleanup System.'' The proposed revision to TS 3/4.7.5 will provide a Limiting Condition of Operation (LCO) 3.0.4 exception for MODES 5, 6, or a defueled configuration.
    Basis for proposed **no** significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of **no** significant hazards consideration, which is presented below:
    1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?
    Response: **No**.
    The Control Room Emergency Air Cleanup System (CREACUS) provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity or toxic gas.
    [The following are the proposed changes to Technical Specification 3/4.7.5 ``Control Room Emergency Air Cleanup System:'']
    Proposed Change 1 [adds the following statement to the Applicability statement of TS 3.7.5: ``or during movement of irradiated fuel assemblies.''] will replace the existing wording of the Applicability with the following words ``ALL MODES or during movement of irradiated fuel assemblies.'' The requirement concerning movement of irradiated fuel assemblies was added because the existing Applicability statement does not reflect the possibility of radiation exposure to the operators inside the control room during this event. A fuel handling accident can happen during defueled operations. In this case, movement of the last irradiated fuel assembly from the empty core inside containment is not covered by the existing Applicability.
    Also, a fuel handling accident can happen inside the Fuel Handling Building when irradiated fuel is moved from one location to another in the Spent Fuel Pool (SEP). The need for the CREACUS during fuel handling is based on safety analysis assumptions which are specified in Chapter 15 of the SONGS Unit 2 and 3 Updated Final Safety Analysis Report (UFSAR).

Addition of the new Applicability requirement will not involve a significant increase in the possibility or consequences of any accident
previously evaluated.

Proposed Change 2 [a new Action d): ``The provisions of Specification 3.0.4 are not applicable when entering MODES 5, 6, or defueled configuration'' is added to the Action section of TS 3.7.5] will add a new Action d) which reads: ``the provisions of Specification
3.0.4 are not applicable when entering MODES 5, 6, or defueled configuration.'' Existing Technical Specification 3/4.7.5 prohibits entering MODE 6 from a defueled configuration unless both CREACUS trains are OPERABLE. With the addition of the statement ``or during movement of irradiated fuel assemblies'' to the Applicability, OPERABILITY of the CREACUS will be ensured prior to movement of irradiated fuel assemblies. Therefore, the only threshold between defueled configuration and MODE 6 is the position of the first irradiated fuel assembly--whether it is in the reactor vessel or external to it. This threshold has **no** safety significance because the only credible event during the transition from a defueled configuration
to MODE 6 and from MODE 6 to defueled configuration is a Design Basis Fuel Handling Accident which is covered by the proposed Applicability. Therefore, this threshold can be expected from Limiting Condition for Operation (LCO) 3.0.4.

The threshold of entering MODE 5 from MODE 6 consists of fully tightening the last reactor vessel head closure bolt. This evolution has **no** safety significance from the point of view of isolating the control room from external hazards. Therefore, this MODE change can be excepted from LCO 3.0.4. The threshold of entering MODE 6 from MODE 5 consists of untightening at least one reactor vessel head closure bolt.
If **no** irradiated fuel assemblies are being moved, this evolution has **no**
safety significance from the point of view of isolating the control room from external hazards. Therefore, this MODE change can be excepted
from LCO 3.0.4 also.

The threshold of entering MODE 5 from MODE 4 consists of decreasing
Reactor Coolant System (RCS) temperature from 350 deg.F > Tavg > 200 deg.F to Tavg [less than or equal to] 200 deg.F by initiating shutdown cooling. If **no** irradiated fuel assemblies are being moved,

this evolution has **no** safety significance from the point of view of isolating the control room from external hazards. Therefore, this MODE change can be excepted from LCO 3.0.4.

The MODE changes have **no** significance relative to releases. Therefore, since CREACUS can be inoperable during each individual mode,
it should not be required to have two OPERABLE CREACUS trains before mode changes.

Therefore, addition of the new Action will not involve a significant increase in the probability or consequences of any accident
previously evaluated.

Proposed Change 3 [adds the following words ``or defueled configuration when moving irradiated fuel assemblies'' after the words ``Units 2 and 3 in MODE 5 or 6'' in the Action statement of TS 3.7.5] will add the words ``or defueled when moving irradiated fuel assemblies'' to the Action statement when either Unit is in MODE 5 or 6. These words are added for consistency with a proposed Applicability statement ``or during movement of irradiated fuel assemblies.'' Without
these words it is not clear what Actions should be entered if the LCO requirement is not met in a defueled configuration when moving irradiated fuel assemblies. By adding these words Actions (a) and (b) became applicable in a defueled configuration when moving irradiated fuel assemblies. This change applies the requirement of the proposed Applicability to the Action when either Unit is in MODES 5 or 6. Therefore, addition of these words to the Action statement will not involve a significant increase in the probability or consequences of any accident previously evaluated.

Proposed Change 4 [adds the following words ``or movement of irradiated fuel assemblies'' after the words ``suspend all operations involving CORE ALTERATIONS or positive reactivity changes'' in the Action (b) statement of TS 3.7.5] will add the words ``or movement of irradiated fuel assemblies'' in the Action (b) statement. These words are added for consistency with the proposed Applicability statement and
proposed Action statement when either Unit is in MODES 5 or 6, or a defueled configuration when moving irradiated fuel assemblies. Without addition of these words Action (b) did not specify what should be done when any Unit is in a defueled configuration when moving irradiated fuel assemblies. Therefore, addition of these words to the Action statement will not involve a significant increase in the probability or

consequences of any accident previously evaluated.
2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any previously evaluated?
Response: **No**.
The changes proposed herein do not reduce the reliability or performance of the Control Room Emergency Air Cleanup System (CREACUS).
The proposed LCO 3.0.4 exception for CREACUS permits MODE 5, MODE 6, or defueled configuration entry with one train of CREACUS inoperable. This change does not affect CREACUS reliability and its capability to perform its intended design functions.
Additional requirements in the Applicability to have two Control Room Emergency Air Cleanup Systems OPERABLE during movement of irradiated fuel covers the consequences of a fuel accident in the Fuel Handling Building and in containment when the reactor vessel is defueled. Operation of the facility will remain unchanged as a result of the proposed changes.
Also, addition of the requirement to suspend movement of irradiated fuel assemblies when either Unit is in a defueled configuration when moving irradiated fuel is made for consistency with the proposed Applicability statement and Action statement. The proposed Action statement emphasize that Actions (a) and (b) are applicable not only when either Unit is in MODES 5 or 6, but also when in a defueled configuration when moving irradiated fuel assemblies. This change does not affect CREACUS reliability and its capability to perform its intended design functions. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?
Response: **No**.
Operation of the facility in accordance with these changes will not be adversely affected as a result of the changes proposed herein. The proposed changes include a change to the Applicability, adding the new Action (d), modifying the Action statement when either Unit is in MODES 5 or 6, and modifying the Action (b). The proposed LCO 3.0.4 exception for CREACUS permits MODE 5, MODE 6, or defueled configuration entry

with one train of CREACUS inoperable. Additional requirements in the Applicability statement to have two Control Room Emergency Air Cleanup Systems OPERABLE during movement of irradiated fuel, covers the consequences of the fuel accident in the Fuel Handling Building. Also, this requirement covers the movement of irradiated fuel when the reactor vessel is defueled. Modified Action statement for either Unit in MODES 5 or 6 is made for consistency with the proposed Applicability statement. Modified Action (b) covers the possibility of both the CREACUS trains being inoperable in a defueled configuration when moving irradiated fuel assemblies.

The margin of safety as defined in Bases 3/4.7.5 is limiting the dose to control room personnel to 5.0 rem or less whole body, or its equivalent. As discussed above, operation of the CREACUS will be unchanged as a result of the proposed changes. Therefore, operation of the facility in accordance with this proposed change will not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Main Library, University of California, P. O. Box 19557, Irvine, California 92713.

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770.

NRC Project Director: Theodore R. Quay.

Southern California Edison Company, et al.

Docket Nos. 50-361 and 50-362.

San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego County, California.

Date of amendment requests: September 16, 1994.

Description of amendment requests: The licensee proposes to revise the linear heat rate (LHR) limit in Technical Specification (TS) 3/4.2.1, ``Linear Heat Rate.'' TS 3/4.2.1 requires maintaining the LHR at or below 13.9 kilowatts per linear foot (kw/ft) for steady-state operation. This amendment request is to revise this value from 13.9 kw/

ft to 13.0 kw/ft. The Bases of TS 3/4.2.1, ``Linear Heat Rate,'' are also being revised to reflect the new value.

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

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

Response: **No**.

The only event impacted by this Technical Specification (TS) change is the Large Break Loss of Coolant Accident (LBLOCA) which has been reanalyzed. There is a direct correlation between the magnitude of the TS 3/4.2.1 Linear Heat Rate (LHR) limit and the calculated peak cladding temperature (PCT). Since the LHR is being reduced in value, which is a conservative change, there will be **no** increase in the consequences of the event. The LBLOCA reanalysis, performed using the new LHR limit in support of an optimized fuel loading pattern, resulted in a reduction of the calculated LBLOCA PCT. Therefore, this change will not involve an increase in the probability or consequences of any accident previously evaluated.

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

Response: **No**.

This amendment request does not involve any change to plant equipment or operation. The linear heat rate limit provided in T/S 3.2.1 is used only in the LBLOCA analysis. **No** change to the LBLOCA methodology was made. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: **No**.

This amendment does not change the manner in which safety limits, limiting safety settings, or limiting conditions for operation are determined. There is **no** change in the PCT acceptance criterion for this event as a result of the proposed reduction in the LHR limit. Therefore, there is **no** reduction in the margin of safety from the acceptance limit to the mechanical failure point of the fuel.

Additionally, the analysis value for the LBLOCA PCT is reduced to 2160 deg.F. This results in an increase in the analysis margin between the acceptance criterion and the analysis value. Therefore, this proposed change does not involve a reduction in a margin of safety.

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

Local Public Document Room location: Main Library, University of California, P.O. Box 19557, Irvine, California 92713.

Attorney for licensee: T.E. Oubre, Esquire, Southern California Edison Company, P.O. Box 800, Rosemead, California 91770.

NRC Project Director: Theodore R. Quay.

Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company

Docket **No**. 50-346

Davis-Besse Nuclear Power Station, Unit **No**. 1, Ottawa County, Ohio.

Date of amendment request: October 7, 1994.

Description of amendment request: The proposed amendment would remove the existing Surveillance Requirement (SR) 4.5.2.d.3 for the Low Pressure Injection (LPI) System and the existing SR 4.6.2.1.c for the Containment Spray (CS) System since the requirement to leak test these systems is programmatically covered in TS 6.8.4.a, ``Primary Coolant Sources Outside Containment.'' Additionally, changes are proposed to TS Bases 3/4.5.2 and 3/4.6.2.1 to reflect the elimination of the above SRs.

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

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

The change does not involve a significant increase in the probability of an accident previously evaluated nor does it involve a significant increase in the consequences of an accident previously evaluated because **no** accident initiators, conditions or assumptions are affected by removing the leak test requirements of LPI System SR

4.5.2.d.3 and CS System SR 4.6.2.1.c. The purpose of these SRs is already encompassed by the existing program requirements of TS 6.8.4.a, ``Primary Coolant Sources Outside Containment.'' TS 6.8.4.a requires integrated leak testing at refueling cycle intervals or less, for each system outside containment, that could contain highly radioactive fluids during a serious transient or accident.

The proposed changes do not alter the source term, containment isolation, or allowable releases. The proposed changes, therefore, will not increase the radiological consequences of a previously evaluated event.

These changes are consistent with NUREG-1430, Revision 0, ``Improved Standard Technical Specifications for B&W Plants.'' The associated changes to TS Bases 3/4.5.2 and 3/4.6.2.1 are administrative.

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

The proposed changes do not create the possibility of any new or different kind of accident from any accident previously evaluated because **no** new accident initiators or assumptions are introduced by these proposed changes to LPI System SR 4.5.2.d.3 and CS System SR 4.6.2.1.c. The purpose of these SRs is already encompassed by the existing program requirements of TS 6.8.4.a, ``Primary Coolant Sources Outside Containment,'' which requires leak testing to be performed on the LPI and CS Systems. These changes are consistent with NUREG-1430. The associated changes to TS Bases 3/4.5.2 and 3/4.6.2.1 are administrative. The proposed changes do not alter any accident scenarios.

(3) The proposed changes do not result in a significant reduction in the margin of safety.

The changes do not involve a significant reduction in the margin of safety because the proposed changes to the LPI System SR 4.5.2.d.3 and CS System SR 4.6.2.1.c do not reduce or adversely affect the capabilities of any plant structures, systems or components. The purpose of these SRs is already encompassed by the existing program requirements of TS 6.8.4.a, ``Primary Coolant Sources Outside Containment.'' These changes are consistent with NUREG-1430. The associated changes to TS Bases 3/4.5.2 and 3/4.6.2.1 are administrative.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c)

are
satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves **no** significant hazards consideration.

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

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

NRC Acting Project Director: C.A. Carpenter.

Virginia Electric and Power Company

Docket Nos. 50-280 and 50-281

Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia.

Date of amendment request: October 11, 1994.

Description of amendment request: The proposed changes would modify
the surveillance frequencies of the containment hydrogen analyzers.

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:

Specifically, operation of Surry Power Station in accordance with the proposed Technical Specifications will not:

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

The proposed changes to the surveillance requirements for the hydrogen analyzers have **no** impact on the probability of any accident occurrence. The hydrogen analyzers are maintained in a standby mode during normal operation and are made fully operable within thirty minutes after a safety injection signal to provide indication of the hydrogen concentration in containment after a loss-of-coolant accident.
This instrumentation is used solely post-accident to monitor containment conditions. Reduced testing of a post-accident monitor does
not contribute to the probability of any previously analyzed accident. These monitors have **no** automatic safety function. Furthermore, the hydrogen analyzers will be operated in the same manner, and the operability requirements are not being altered. In addition, the Post-Accident Sampling System serves as a diverse means to confirm post-accident hydrogen concentration in containment. Therefore, the consequences of a Design Basis Accident are not being increased by the

proposed change in surveillance test frequency of the hydrogen analyzers.

Reducing the frequency of surveillance testing could however decrease the timeliness in identifying an inoperable hydrogen analyzer.
However, our surveillance test experience has shown that the analyzers have been very stable with repeatable results, and we conclude that the
change in test frequency should not affect the reliability or operability of the analyzers. Likewise, the NRC has determined in Generic Letter 93-05 that a reduced frequency of surveillance testing during power is acceptable to determine hydrogen analyzer operability.

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

There are **no** plant modifications or changes in methods of plant operation introduced by this change in hydrogen analyzer surveillance frequencies. The hydrogen analyzers are maintained in a standby mode during normal operation and are fully operable within thirty minutes after a safety injection signal to provide indication of the hydrogen concentration in containment after a loss-of-coolant accident. Therefore, the possibility of a new or different kind of accident than previously evaluated is not created by the proposed changes in surveillance frequency of the control rods [hydrogen analyzers surveillance frequencies].

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

The hydrogen analyzer surveillance requirements do not affect the margin of safety in that the operability requirements for the safety systems and containment remain unchanged. The hydrogen analyzers only provide indication and do not perform any direct function to mitigate the consequences of any previously analyzed accidents. Furthermore, the
change in surveillance frequency is associated with a post-accident monitor with **no** automatic safety functions and a diverse means of confirming the parameter by the Post-Accident Sampling System. Therefore, the margin of safety is not altered by this proposed change in the surveillance frequencies of the hydrogen analyzers.

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

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

Attorney for licensee: Michael W. Maupin, Esq., Hunton and

Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.
    NRC Project Director: Mohan C. Thadani (Acting).

Previously Published Notices of Consideration of Issuance of Amendments
to Facility Operating Licenses, Proposed **No** Significant Hazards Consideration Determination, and Opportunity for a Hearing

    The following notices were previously published as separate individual notices. The notice content was the same as above. They were
published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving **no** significant hazards consideration.
    For details, see the individual notice in the Federal Register on the day and page cited. This notice does not extend the notice period of the original notice.

Northeast Nuclear Energy Company, et al.

Docket **No**. 50-336

    Millstone Nuclear Power Station, Unit **No**. 2, New London County, Connecticut.
    Date of amendment request: September 26, 1994.
    Description of amendment request: The proposed amendment would revise the Technical Specifications by adding a footnote to Surveillance Requirement 4.6.1.2.d that defers the performance of Type B and C containment leak rate tests to the end of the twelfth refueling
outage.
    Date of publication of individual notice in Federal Register: October 13, 1994, (59 FR 52005).
    Expiration date of individual notice: November 14, 1994.
    Local Public Document Room location: Learning Resource Center, Three Rivers Community--Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 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 Nos. STN 50-528, STN 50-529, and STN 50-530

Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona.

Date of application for amendments: August 23, 1993, as supplemented by letter of July 21, 1994.

Brief description of amendments: These amendments remove the Units 1 and 3 license condition regarding an augmented reactor coolant pump vibration monitoring program and the confirmatory order modifying the Unit 2 license regarding the same issue.

Date of issuance: October 27, 1994.

Effective date: October 27, 1994.
Amendment Nos.: 84, 72, and 56.
Facility Operating License Nos. NPF-41, NPF-51, and NPF-74: The amendments revised the Technical Specifications.
Date of initial notice in Federal Register: September 29, 1993 (58 FR 50963).
The additional information in the letter dated July 21, 1994, was clarifying in nature and did not affect the staff's previously published **no** significant hazards determination.
The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 27, 1994.
No significant hazards consideration comments received: **No**.
Local Public Document Room location: Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004

Baltimore Gas and Electric Company

Docket Nos. 50-317 and 50-318

Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 & 2, Calvert County, Maryland.
Date of application for amendments: August 4, 1994.
Brief description of amendments: The amendments delete Technical Specifications 3/4.3.3.3, 6.9.2.b, 6.9.2.d, and Bases 3/4.3.3.3, which provide the requirements for the operation and the testing of seismic monitoring instrumentation, and relocates them to the Updated Final Safety Analysis Report and plant procedures.
Date of issuance: October 21, 1994.
Effective date: As of the date of issuance to be implemented within
30 days.
Amendment Nos.: 199 and 176
Facility Operating License **No**. DPR-**53** and DPR-69: Amendment revised
the Technical Specifications.
Date of initial notice in Federal Register: September 14, 1994 (59 FR 47165).
The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated October 21, 1994.
No significant hazards consideration comments received: **No**.
Local Public Document Room location: Calvert County Library, Prince
Frederick, Maryland 20678.

Baltimore Gas and Electric Company

Docket Nos. 50-317 and 50-318

Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland.

Date of application for amendments: November 4, 1993.

Brief description of amendments: These amendments revise the Updated Final Safety Analysis Report to address the removal of orifice plates in the containment vent/purge lines of each unit and revise the maximum hypothetical accident analysis to address the increased flow as the result of removing the orifice plates.

Date of issuance: October 21, 1994.

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

Amendment Nos.: 200 and 177.

Facility Operating License **No**. DPR-**53** and DPR-69: Amendment revised the Licenses.

Date of initial notice in Federal Register: February 25, 1994 (59 FR 9254).

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated October 21, 1994.

No significant hazards consideration comments received: **No**.

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

Commonwealth Edison Company

Docket Nos. STN 50-454 and STN 50-455

Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois.

Date of application for amendments: August 1, 1994, as supplemented by your letters dated September 7, 1994, and September 17, 1994 (two letters), with clarifying information submitted by letters dated September 22, 1994, September 23, 1994, September 30, 1994, October 17, 1994, and October 24, 1994.

Brief description of amendments: The purpose of the amendment is to
incorporate voltage-based repair criteria into the Byron, Unit 1, technical specifications, thereby permitting the use of voltage-based steam generator (SG) tube plugging criteria for a specific class of SG tube defects.

Date of issuance: October 24, 1994.

Effective date: October 24, 1994.

Amendment Nos.: 66 and 66.

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

Date of initial notice in Federal Register: September 23, 1994 (59 FR 48917).

The clarifying information in the September 22, 1994, September 23,
1994, September 30, 1994, October 17, 1994, and October 24, 1994, submittals did not affect the initial determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 24, 1994.

No significant hazards consideration comments received: **No**.

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

Commonwealth Edison Company

Docket Nos. STN 50-454 and STN 50-455

Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois.

Docket Nos. STN 50-456 and STN 50-457

Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois.

Date of application for amendments: March 23, 1994, as supplemented
on July 26, 1994.

Brief description of amendments: The amendments change the Technical Specifications to reflect a reduced thermal flow to compensate for increased steam generator tube plugging up to 15 percent
of the total number of tubes. The amendment also approves the use of higher boron concentration in the refueling water storage tank, the reactor coolant system accumulators, and the refueling cavity.

Date of issuance: October 21, 1994.

Effective date: October 21, 1994.
Amendment Nos.: 65, 65, 56, and 55.
Facility Operating License Nos. NPF-37, NPF-66, NPF-72 and NPF-77: The amendments revised the Technical Specifications.
Date of initial notice in Federal Register: August 15, 1994 (59 FR 41802).
The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 21, 1994.
No significant hazards consideration comments received: **No**.
Local Public Document Room location: For Byron, the Byron Public Library, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Township Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Consumers Power Company

Docket **No**. 50-255

Palisades Plant, Van Buren County, Michigan.
Date of application for amendment: November 15, 1991, supplemented February 22, March 11, April 7, and August 23, 1994.
Brief description of amendment: This amendment is a complete rewrite of the instrumentation operability requirements.
Date of issuance: October 26, 1994.
Effective date: October 26, 1994.
Amendment **No**.: 162.
Facility Operating License **No**. DPR-20. Amendment revised the Technical Specifications.
Date of initial notice in Federal Register: May 25, 1994 (59 FR 27052)
The August 23, 1994, request contained editorial changes within the
scope of the initial notice and did not affect the staff's proposed **no** significant hazards consideration findings. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 26, 1994.
No significant hazards consideration comments received: **No**.
Local Public Document Room location: Van Wylen Library, Hope College, Holland, Michigan 49423.

Entergy Operations, Inc.,

Docket **No**. 50-313

Arkansas Nuclear One, Unit **No**. 1, Pope County, Arkansas.
Date of amendment request: January 13, 1994.
Brief description of amendment: The amendment revised the specifications governing the reactor protection system (RPS) to permit the plant to operate indefinitely with one RPS channel in by-pass.
Date of issuance: October 24, 1994.
Effective date: October 24, 1994.
Amendment **No**.: 174.
Facility Operating License **No**. DPR-51. Amendment revised the Technical Specifications.
Date of initial notice in Federal Register: March 2, 1994 (59 FR 10005).
The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 24, 1994.
No significant hazards consideration comments received: **No**.
Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801.

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: February 18, 1994, as supplemented by letter dated August 5, 1994.
Brief description of amendments: These amendments delete the audit frequencies from the Technical Specifications (TS) and modify the TS administrative control requirements for emergency and security plans.
Date of issuance: October 26, 1994.
Effective date: October 26, 1994.
Amendment Nos: 168 and 162.
Facility Operating Licenses Nos. DPR-31 and DPR-41: Amendments revised the Technical Specifications.
Date of initial notice in Federal Register: March 30, 1994 (59 FR 14889). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 26, 1994.
No significant hazards consideration comments received: **No**.
Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199.

GPU Nuclear Corporation, et al.

Docket **No**. 50-219

Oyster Creek Nuclear Generating Station, Ocean County, New Jersey.

Date of application for amendment: August 19, 1994.

Brief description of amendment: The amendment updates and clarifies the surveillance requirements for control rod exercising and standby liquid control pump operability testing to be consistent with Generic Letter 93-05.

Date of Issuance: October 19, 1994.

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

Amendment **No**.: 172.

Facility Operating License **No**. DPR-16. Amendment revised the Technical Specifications.

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

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated October 19, 1994.

No significant hazards consideration comments received: **No**.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

IES Utilities Inc.

Docket **No**. 50-331

Duane Arnold Energy Center, Linn County, Iowa.

Date of application for amendment: May 28, 1992, as supplemented on January 6, May 27 and October 20, 1994.

Brief description of amendment: The amendment revised the Technical Specifications by changing the limiting conditions for operation and surveillance requirements for primary containment integrity, secondary containment integrity, and other systems and equipment of Section 3.7, Containment Systems. Limiting conditions for operation and surveillance requirements for drywell average air temperature and secondary containment automatic isolation dampers were also added.

Date of issuance: October 26, 1994.

Effective date: October 26, 1994, to implemented within 120 days.

Amendment **No**.: 201

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

Date of initial notice in Federal Register: July 6, 1994 (59 FR 34665). The licensee's October 20, 1994, submittal, provided clarifying information at the request of the NRC staff. This submittal did not change the initial application or the **no** significant hazards determination as originally noticed. Therefore, renoticing was not warranted.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 26, 1994.

No significant hazards consideration comments received: **No**.

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

Northeast Nuclear Energy Company, et al.

Docket **No**. 50-423

Millstone Nuclear Power Station, Unit **No**. 3, New London County, Connecticut.

Date of application for amendment: September 30, 1993, as supplemented July 8, 1994.

Brief description of amendment: The amendment revises the Technical Specifications by increasing the minimum volume of fuel oil required to be stored in the emergency diesel generator (EDG) day tank from 205 gallons to 278 gallons, and clarifies the bases for the EDG fuel oil storage tank and day tank minimum fuel oil volume requirements.

Date of issuance: October 17, 1994.

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

Amendment **No**.: 97.

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

Date of initial notice in Federal Register: November 10, 1993 (58 FR 59753).

The July 8, 1994, letter provided clarifying information that did not change the initial proposed **no** significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 17, 1994.

No significant hazards consideration comments received: **No**.

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

Pennsylvania Power and Light Company

Docket Nos. 50-387 and 50-388

Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania.

Date of application for amendments: May 31, 1994.

Brief description of amendments: These amendments change the frequency for monitoring the Susquehanna site spray pond ground water level from once per month to once every 6 months.

Date of issuance: October 20, 1994.

Effective date: Both units; as of date of issuance and to be implemented within 30 days after the date of issuance.

Amendment Nos.: 135 and 105.

Facility Operating License Nos. NPF-14 and NPF-22. These amendments revised the Technical Specifications.

Date of initial notice in Federal Register: July 6, 1994 (59 FR 34668).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 20, 1994.

No significant hazards consideration comments received: **No**.

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

Philadelphia Electric Company, Public Service Electric and Gas Company Delmarva Power and Light Company, and Atlantic City Electric Company

Docket **No**. 50-277

Peach Bottom Atomic Power Station, Unit **No**. 2, York County, Pennsylvania.

Date of application for amendment: June 23, 1993, as supplemented by letters dated April 5, May 2, June 6, June 8, July 6 (two letters), July 7, July 20, July 28, 1994 (two letters), September 16, September 30, and October 14, 1994. The supplemental letters provided clarifying information that did not change the initial proposed **no** significant hazards consideration determination.

Brief description of amendment: The amendment raises the authorized maximum power level from 3293 MWt to a new limit of 3458 MWt.

Date of issuance: October 18, 1994.

Effective date: Unit 2, effective as of its date of issuance and is to be implemented prior to startup in Cycle 11 currently scheduled for October 28, 1994.

Amendment **No**.: 198.

Facility Operating License **No**. DPR-44: The amendment revised the license and Technical Specifications.

Date of initial notice in Federal Register: August 29, 1994 (59 FR 44432).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 18, 1994.

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 Nos. 50-272 and 50-311

Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey.

Date of application for amendments: February 18, as supplemented by letter dated April 6, 1994 for Salem Unit 1 and March 28, 1994 for Salem Unit 2.

Brief description of amendments: The change to Salem Unit 1 Technical Specifications (TS) replaces the main feedwater control and control bypass valves with the main feedwater stop check valves for the Containment Isolation Function. The change to Salem Unit 2 TS adds a footnote to the 21-24 BF22 (main feedwater stop check valves) on Table 3.6-1, ``Containment Isolation Valves.'' This note identifies those

containment isolation Valves that are not subject to 10 CFR Part 50,
Appendix J, Type C leakage testing.
    Date of issuance: October 20, 1994.
    Effective date: Units 1 and 2, effective as of date of issuance and
shall be implemented within 60 days of the date of issuance.
    Amendment Nos.: 158 and 139.
    Facility Operating License Nos. DPR-70 and DPR-75. The amendments
revised the Technical Specifications.
    Date of initial notice in Federal Register: July 20, 1994 (59 FR 37083).
    The Commission's related evaluation of the amendments is contained
in a Safety Evaluation dated October 20, 1994.
    No significant hazards consideration comments received: **No**.
    Local Public Document Room location: Salem Free Public Library, 112
West Broadway, Salem, New Jersey 08079.

Southern California Edison Company, et al.

Docket Nos. 50-361 and 50-362

    San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego
County, California
    Date of application for amendments: December 31, 1992.
    Brief description of amendments: These amendments revise the
Technical Specifications (TS) to (1) distinguish between the core
operating limit supervisory system (COLSS) in service and the COLSS out
of service (OOS), (2) add surveillances to monitor departure from
nucleate boiling ratio (DNBR) and/or linear heat rate (LHR) every 15
minutes when the COLSS is OOS and the corresponding parameter is not
being maintained as required, (3) increase the ACTION time from 1 hour
to 4 hours when the COLSS is OOS and either the LHR or DNBR margin is
not being maintained within the required limits, (4) change the power
reduction requirements from ``HOT STANDBY'' to ``less than or equal to
20 percent of Rated Thermal Power'' when the DNBR margin and/or the LHR
limiting condition for operation (LCO) cannot be met within the allowed
ACTION time, and (5) revise the Bases to the TS to reflect these
changes.

Date of issuance: October 27, 1994.
Effective date: October 27, 1994.
Amendment Nos.: 113 and 102.
Facility Operating License Nos. NPF-10 and NPF-15: The amendments revised the Technical Specifications.
Date of initial notice in Federal Register: March 3, 1993 (58 FR 12269).
The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 27, 1994.
No significant hazards consideration comments received: **No**.
Local Public Document Room location: Main Library, University of California, P.O. Box 19557, Irvine, California 92713.

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: May 18, 1994; revised September 9, 1994 (TS 94-05).
Brief description of amendments: The amendments revise the action statement to provide a fixed duration that the control room emergency ventilation system may be inoperable due to actions taken as a result of a tornado warning.
Date of issuance: October 17, 1994.
Effective date: October 17, 1994.
Amendment Nos.: 187 and 179.
Facility Operating License Nos. DPR-77 and DPR-79: Amendments revise the technical specifications.
Date of initial notice in Federal Register: June 22, 1994 (59 FR 32237).
The Commission's related evaluation of the amendments are contained in a Safety Evaluation dated October 17, 1994.
No significant hazards consideration comments received: None.
Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

Tennessee Valley Authority

Docket Nos. 50-327 and 50-328

Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee.
Date of application for amendments: August 19, 1994 (TS 93-09).
Brief description of amendments: The amendments delay implementation of Amendments Nos. 182 and 174 from the Unit 2 Cycle 6 refueling outage to as soon as acceptable plant conditions and modification activities/procedures are established in fiscal year 1995.
Date of issuance: October 17, 1994.
Effective date: October 17, 1994.
Amendment Nos.: 188 and 180.
Facility Operating License Nos. DPR-77 and DPR-79: Amendments revise the technical specifications.
Date of initial notice in Federal Register: September 14, 1994 (59 FR 47182).
The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 17, 1994.
No significant hazards consideration comments received: None.
Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.
No significant hazards consideration comments received: None.
Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

Tennessee Valley Authority

Docket Nos. 50-327 and 50-328

Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee.
Date of application for amendments: September 8, 1994 (TS 94-14).
Brief description of amendments: The amendments incorporate clarifications regarding the evaluation of steam generator tube defects by separating the portion of the steam generator tube starting at the end of the tube up to the start of the tube-to-tube sheet weld from the remainder of the tube for the purposes of sample selection and repair when defects are found in this section of a steam generator tube.
Date of issuance: October 20, 1994.
Effective date: October 20, 1994.
Amendment Nos.: 189 and 181.
Facility Operating License Nos. DPR-77 and DPR-79: Amendments revise the technical specifications.
Date of initial notice in Federal Register: September 19, 1994. (59

FR 47962).
The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 20, 1994.
No significant hazards consideration comments received: None.
Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

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: April 5, 1994.
Brief description of amendment: The amendment increases the surveillance test interval for the turbine-driven auxiliary feedwater pump and motor-driven feedwater pump from 31 days to 92 days; clarifies
a requirement for a dedicated individual to be stationed at manual valves during surveillance testing because of the availability of the motor-driven feedwater system; addresses miscellaneous editorial corrections, and revises TS 3/4.7.1.2 and TS 3/4.1.7 and their associated bases.
Date of issuance: October 18, 1994.
Effective date: October 18, 1994.
Amendment **No**.: 193.
Facility Operating License **No**. NPF-3: Amendment revised the Technical Specifications.
Date of initial notice in Federal Register: May 25, 1994 (59 FR 27068).
The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 18, 1994.
No significant hazards consideration comments received: **No**.
Local Public Document Room location: University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606.

Union Electric Company

Docket **No**. 50-483

Callaway Plant, Unit 1, Callaway County, Missouri.
Date of application for amendment: February 10, 1994.
Brief description of amendment: The amendment revises the

Technical
Specification Table 2.2-1, ``Reactor Trip System Instrumentation Trip Setpoints,'' to correct Total Allowance values. The associated Bases section clarifies the relationship between the power supply and undervoltage relays.

Date of issuance: October 27, 1994.

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

Amendment **No**.: 93.

Facility Operating License **No**. NPF-30. Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: March 30, 1994 (59 FR 14897).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 27, 1994.

No Significant hazards consideration comments received: **No**.

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

Vermont Yankee Nuclear Power Corporation

Docket **No**. 50-271

Vermont Yankee Nuclear Power Station, Vernon, Vermont.

Date of application for amendment: December 6, 1993.

Brief description of amendment: The proposed change removes the requirement to perform jet pump integrity and operability surveillances in the idle loop during operation with one recirculation loop.

Date of issuance: October 26, 1994.

Effective date: October 26, 1994.

Amendment **No**.: 141.

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

Date of initial notice in Federal Register: June 8, 1994 (59 FR 29637).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 26, 1994.

No significant hazards consideration comments received: **No**.

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

Virginia Electric and Power Company

Docket Nos. 50-280 and 50-281

Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia.

Date of application for amendments: October 19, 1993.

Brief description of amendments: These amendments will add operability requirements, action statements, and surveillance requirements for the recirculation spray heat exchanger service water outlet radiation monitors. Also, surveillance requirements for several post-accident instruments are being reinstated.

Date of issuance: October 27, 1994.

Effective date: October 27, 1994.

Amendment Nos.: 193 and 193.

Facility Operating License Nos. DPR-32 and DPR-37: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: December 22, 1993 (58 FR 67864).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 27, 1994.

No significant hazards consideration comments received: **No**.

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

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 December 9, 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).

Wisconsin Electric Power Company

Docket Nos. 50-266 and 50-301

Point Beach Nuclear Plant, Unit Nos. 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin.

Date of application for amendments: October 20, 1994.

Brief description of amendments: These amendments revise Technical Specification (TS) Section 15.3.1.G, ``Operational Limitations,'' to reduce the reactor coolant system raw measured total flow rate and operating pressure, modify TS Section 15.2.3.1.B to increase the required reduction in the delta-T trip setpoint, and modify TS Figure 15.2.1-1 to reflect new reactor core safety limits, all for Unit 2 only. The applicable bases are also revised.

Date of issuance: October 28, 1994.

Effective date: October 28, 1994.

Amendment Nos.: 156 and 160.

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

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

The Commission's related evaluation of the amendments, finding of emergency circumstances, and final determination of **no** significant hazards consideration are contained in a Safety Evaluation dated October 28, 1994.

Local Public Document Room location: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

Attorney for licensee: Ernest L. Blake, Jr., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

Acting NRC Project Director: Cynthia A. Carpenter.

Dated at Rockville, Maryland, this 2nd day of November, 1994.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 94-27613 Filed 11-8-94; 8:45 am]
BILLING CODE 7590-01-P