significant hazards consideration in accordance with 10 CFR 50.91 and 50.92.

For further details with respect to this action, see the application for amendments dated September 6, 1996, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC. and at the local public document room located at the Athens Public Library, 405 E. South Street, Athens, Alabama.

Dated at Rockville, Maryland, this 17th day of October 1996.

For the Nuclear Regulatory Commission.

Frederick J. Hebdon,

Director, Project Directorate II–3, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 96–27162 Filed 10–22–96; 8:45 am]

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UNITED STATES NUCLEAR REGULATORY COMMISSION

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from September 30, 1996, through October 10, 1996. The last biweekly notice was published on October 9, 1996 (61 FR 52962).

Notice Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

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

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

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

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

By November 22, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (Project Director): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the attorney for the licensee.

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

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

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

Date of amendment request: September 18, 1996

Description of amendment request: Revise Technical Specification (TS) 4.8.1.1.2 by removing TS 4.8.1.1.2.h.2 pressure testing requirement since adequate testing will be completed in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI.

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:

This change does not involve a significant hazards consideration for the following reasons:

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

Applying ASME Code, Section XI alternative examination/testing will not affect any initiators of any previously evaluated accidents or change the manner in which the emergency diesel generators or any other systems operate. The diesel fuel oil system supports the emergency diesel generators which serve an accident mitigating function. Where portions of piping are non-isolable or where atmospheric tanks are involved, the Section XI ASME alternatives to 110% pressure testing continue to ensure the integrity of the fuel oil system without any impact on analyzed accident scenarios or their consequences. Therefore, the proposed amendment does not result in an 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 proposed alternative testing and surveillance will not involve any physical alterations or additions to plant equipment or alter the manner in which any safety-related system performs it function. Using ASME Section XI, or NRC-approved ASME Code cases, as guidance for pressure testing continues to provide assurance that the fuel oil supply system will perform its intended function. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

There are no changes being made to the safety limits or safety settings that would adversely impact plant safety. Further, there is no impact on the margin of safety as defined in the Technical Specifications. Utilizing ASME Section XI as guidance for determining those sections of piping that should be pressure-tested or tested at atmospheric pressure will ensure proper operation of the diesel generator fuel oil supply system. 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: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605

Attorney for licensee: William D. Johnson, Vice President and Senior Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: F. Mark Reinhart, Acting

Detroit Edison Company, Docket No. 50-16, Enrico Fermi Atomic Power Plant, Unit 1, Monroe County, Michigan

Date of amendment request: August 29, 1996 (Reference NRC-96-0111)

Description of amendment request: The proposed amendment will: (1) allow certain equipment and instruments to be removed from service for short periods of time to allow for maintenance, testing, inspection, modifications, and account for equipment failures; (2) reduce the frequency of environmental liquid effluent monitoring and eliminate one raw water sampling location; (3) eliminate the requirement for moisture intrusion monitoring for the reactor building lower level; and (4) correction of a typographical error.

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 using the standards in 10 CFR 50.92(c). The licensee's analysis is presented below:

(1) The operation of Enrico Fermi Atomic Power Plant, Unit 1, in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not involve a significant increase in the probability or consequences of an accident. Provisions for removing the primary cover gas supply from service for short periods of time will not significantly increase the probability of an accident occurring as long as the probability of a significant water reaction with residual sodium is not significantly increased. This is ensured by prescribing limits on the time that carbon dioxide pressure can be low. The consequences of an accident would not be affected by provisions for removing the primary cover gas supply from service as this equipment does not mitigate accidents or affect the accident sequences. Similarly, the provisions for removing the moisture intrusion and cover gas pressure alarms from service for short period of time will not significantly increase the probability of an accident. The alarms provide a monitoring function to detect degradation in the performance of the cover gas supply and sump systems. Absence of these alarm functions for short periods of time does not increase the probability of such degradation and it does not significantly impact the ability for timely detection of such degradation. The consequences of an accident would not be affected by provisions for removing the moisture intrusion and cover gas pressure alarms from service as this equipment does not mitigate accidents or affect the accident sequences. Elimination of the moisture intrusion alarm for the reactor building lower level does not significantly increase the probability of an accident because the probability that water could accumulate in this area is essentially unchanged. Design features of the foundation, containment structure, and annulus drains are intended to prevent entry of water into the reactor building. These features have prevented any water intrusion into this area. The consequences of an accident would not be affected by elimination of the moisture intrusion alarm for the reactor building lower level because this equipment does not mitigate accidents or affect the accident sequences. The Safety Evaluation Supporting Amendment 9 to the referenced license did not rely on moisture intrusion monitoring and alarm features for any safety function or accident prevention or mitigation function. Environmental monitoring surveillance are unrelated to postulated accident sequences and cannot affect the probability or consequences of an accident. The correction of the typographical error is unrelated to accident initiation and sequences and cannot affect the probability or consequences of any accident.

(2) The operation of Enrico Fermi Atomic Power Plant, Unit 1, in accordance with the proposed amendment will 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 a new or different accident from any previously evaluated. With the exception of the allowance for composite environmental samples, which are unrelated to any potential accident sequence, these changes propose no new activities or new methods for performing existing activities. Previous evaluations have considered the release of all of the radioactivity in the residual sodium due to postulated fire or other catastrophe and release of radioactive water stored in the liquid waste tanks which bound the only possible radiological accidents at Fermi 1. For these reasons, no new or different type of accident is created by these changes.

(3) The operation of Enrico Fermi Atomic Power Plant, Unit 1, in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

The proposed changes do not involve a significant reduction in a margin of safety. The changes to the primary system cover gas system technical specifications still ensure that any residual sodium is passivated by carbon dioxide. Changes to the alarms affect only monitoring functions and therefore do not cause a change to any parameter that could affect the margin of safety. Similarly, the environmental surveillances are unrelated to margin of safety. The correction of the typographical error is unrelated to margin of safety. For these reasons, the proposed changes do not involve a significant reduction in the margin of safety.

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

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

Attorney for licensee: John Flynn, Esquire, Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226NRC Branch Chief: Michael F. Weber

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

Date of amendment request: September 25, 1996 (NRC-96-0085)

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Surveillance Requirement 4.8.4.3 to remove the requirement to periodically test the thermal overload (TOL) devices for safety-related motor-operated valves (MOVs). The surveillance requirement would continue to require testing of a TOL device following any maintenance activity that could affect the performance of the device. The surveillance requirement would also be clarified by indicating that testing of TOL devices is required upon initial installation. The associated portion of the TS Bases would also be revised to reflect this change.

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

1. The proposed change does not involve a significant increase in the probability or consequences of an accident. The deletion of the requirement for testing of the TOL protective devices lessens degradation to the components which can improve MOV reliability. Based on historical data through the years of testing, there is no significant drifting of the trip setpoints of the TOL protective devices. The probability of an accident would not increase since terminating the periodic testing or clarifying the situational testing requirements cannot cause equipment to operate inadvertently and so cannot cause an accident. The periodic testing of the TOL protective devices can temporarily render MOVs inoperable due to the removal of the components from service and can cause safety systems/divisions to become unavailable. The deletion of the periodic testing requirement would increase the availability of safety systems insuring that they would be able to respond to accident conditions. The consequences of an accident will not increase since eliminating the periodic testing and clarifying the situational testing requirements will improve reliability of safety-related MOVs to respond to an accident and will not increase the failure rate of equipment. The clarification of the situational testing ensures that the test will be conducted after any maintenance that could affect the performance of the TOL protective devices. Thus, the proposed change increases reliability of the MOVs and increases plant safety. Therefore this change will not result in a significant increase in the probability or consequences of an accident.

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

protective devices are not an accident initiator, they only protect equipment provided to mitigate the consequences of an accident. For this reason, no new or different type of accident is created by this change.

3. The proposed change does not involve a significant reduction in a margin of safety. The trip setpoints of the TOL protective devices depend upon both the current and the length of time the current is applied. The trip setpoints for TOL protective devices are much higher than conditions normally experienced during an MOV stroke and are meant to protect the motor from stall and overload conditions. The difference between the current of the trip setpoints and the normal conditions is great enough that a premature trip of the TOL protective device is highly unlikely, even at degraded voltages. The TOL protective device protects the motor from the stall conditions. Not conducting the periodic testing of the TOL protective devices would not cause the MOVs to fail, nor would the performance of the MOVs be adversely affected. Throughout the life of the plant, there has never been an instance of a safety related MOV failure due to degradation or failure of TOL protective devices. Further, based on maintenance history, the elimination of the periodic testing would eliminate any significant potential degradation of the TOL protective devices, thereby increasing their reliability. Finally, with the removal of the periodic testing of the TOL protective devices, fewer MOVs would have to be removed from service for testing. Since necessary components would no longer be inoperable due to the periodic testing, there would be an increase of availability time of safety systems/divisions. Deletion of the periodic testing could reduce the durations of online system outages. Clarifying the situational testing requirements would better define when the testing of the TOL protective devices is necessary which would ensure operability. The testing would be based on installation or any maintenance that could affect the TOL protective device. For these reasons, the proposed change does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: John N. Hannon

Duke Power Company, Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: June 21, 1996

Description of amendment request: The proposed amendments would administratively correct the term ''lifting load'' in Technical Specification 3.9.6b.2 to ''lifting force.'' This correction would clarify that the static loads associated with the lifting tool, drive rod and control rod weights are not included in the lifting force limit. The amendments would also more accurately define auxiliary hoist minimum capacities and give a more expansive description of the activities for which protective measures and surveillance testing are used.

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:

Question: Will the change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed change[s] [are] administrative in nature, and do[] not represent any changes to the refueling process in the field. It more accurately describes the components for which the LCO's [limiting conditions of operation] protection is intended as well as giving a more accurate description of the auxiliary hoist's minimum capacity. [They] also broaden[] the domain of activities for which protective measures are taken, by including drag load testing into monitored activities. At both MNS [McGuire Nuclear Station] and CNS [Catawba Nuclear Station], the auxiliary hoists and the manipulator cranes are rated at [greater than or equal to] 3000 pounds and are surveillance tested to greater than 1000 pounds. This brackets the limit force lifting value change from 600 to 1000 pounds in the amendment proposal.

Question: Will the change create the possibility of a new or different kind of accident from any accident previously evaluated?

No. Th[ese] proposed administrative change[s] reflect[] no changes in the refueling processes, or any systems, structures or components connected with the refueling process.

Question: Will the change involve a significant reduction in a margin of safety?

No. The proposed administrative change[s] [have] no impact on refueling processes, systems, structures or components, and do[] not result in any significant reduction in a margin of safety. The subject change[s] only clarif[y] the original intent of the specification and more accurately describe[] the involved components, component capacities and the domain of activities for which measures are taken to protect the reactor internals.

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 proposed amendments involve no significant hazards consideration.

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

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

NRC Project Director: Herbert N. Berkow

Duke Power Company, Docket Nos. 50-269, 50-270 and 50-287, Oconee Nuclear Station, Units 1, 2 and 3, Oconee County, South Carolina

Date of amendment request: September 17, 1996 (TSC 96-01)

Description of amendment request: The proposed changes would reduce the Reactor Building pressure setpoint for actuation of the Reactor Building Spray System in Technical Specification (TS) 3.5.3 from a maximum of 30 pounds per square inch gauge (psig) to 15 psig, reduce the maximum allowable Reactor Building internal pressure specified in TS 3.6.4 from 1.5 psig to 1.2 psig when the reactor is critical, revise the corresponding Bases of TS 3.3 to indicate that the Reactor Building sprays and coolers are designed to mitigate the containment temperature response rather than containment pressure response to a loss-of-coolant accident, and make other administrative changes. In addition, the lower Reactor Building pressure limit (a vacuum of 5 inches of mercury (Hg)) in Specification 3.6.4 would be changed to the corresponding value in terms of psig to reflect the units displayed on the control room instrumentation.

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

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

No. The analysis of the post-LOCA [loss-of-coolant accident] Reactor Building response to high-energy line breaks, using the new methodology, uses assumptions different from the requirements currently delineated in Technical Specifications. The new assumptions used for initial Reactor Building pressure and Reactor Building Spray system actuation are 1.2 psig and 20 psig respectively. These values are lower, and hence more conservative, than the values currently specified in Technical Specifications.

Since the new values for Reactor Building pressure and Reactor Building Spray actuation are more conservative and the analysis methodology has received approval from the NRC via [an] SER, this change does not involve a significant increase in the probability or consequences of an accident previously identified.

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

No. The methodology for Reactor Building high energy line break analysis is being revised. The revision of the method of analysis does not alter the manner by which plant systems and components function for accident mitigation.

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

No. By letter dated March 15, 1995, the NRC stated that the new analyses described in the topical report, DPC-NE-3003-P, expand the scope of analyzed piping failures in containment for the Oconee facilities. The NRC further stated that this new analysis method has been used to reanalyze existing licensing basis pipe failure events in containment, and to examine the potential effects of previously unanalyzed assumptions and initial conditions which the NRC staff finds to be consistent with current NRC staff acceptance criteria or produce equally conservative results. In conclusion, the NRC confirmed that this methodology, with appropriate adjustments to reflect potential plant modifications, may be used by Duke Power to perform future analyses in support of licensing applications related to containment accident response. This proposed change to Technical Specifications reflects the use of this new methodology. Based on this new methodology, changes have been made to setpoint assumptions for initial Reactor Building pressure and Reactor Building Spray actuation. This proposed Technical Specification change reflects those assumption changes. This methodology has been accepted by the NRC. This proposed change to Technical Specifications does not involve a significant reduction in the margin of safety.

The NRC has reviewed the licensee's analysis and, based on thisreview, 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: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691

Attorney for licensee: J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20036

NRC Project Director: Herbert N. Berkow

Duquesne Light Company, et al., Docket No. 50-334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of amendment request: September 9, 1996

Description of amendment request: The proposed amendment would revise the Minimum Channels Operable requirement of Item 4.c (Steam Line Isolation, Containment Pressure Intermediate -- High-High) of Technical Specification (TS) Table 3.3-3 from 3 to 2. This proposed change would make this Unit 1 TS consistent with the comparable Unit 2 TS.

The proposed amendment would also revise the minimum charging pump discharge pressure in TS 3.5.5 from 2311 psig to 2397 psig. This change is required to ensure that safety analysis assumptions for safety injection flow are met. Conforming changes would also be made to the Bases for TS 3/4.5.5 to reflect the proposed changes to TS 3.5.5.

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

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

The proposed amendment does not add or modify any existing plant equipment. Since normal charging pump discharge pressure is greater than or equal to approximately 2440 psig, no additional plant configuration changes or modifications will be required to comply with this revised charging pump discharge pressure value. The proposed amendment does not change the design or function of the containment pressure intermediate-high-high channels.

The consequences of an accident previously evaluated are not significantly increased. The ability of the containment pressure intermediate-high-high function to initiate steam line isolation will not be affected. Since steam line isolation will continue to occur at the same required trip setpoint, the amount of mass and energy released to containment along with the ability to maintain at least one unfaulted steam generator (SG) as a heat sink for the reactor remains unchanged. The amount of seal injection flow will continue to be adequately limited to ensure sufficient flow to the reactor core during accident conditions. The Bases changes are editorial in nature and do not involve a change to probability or consequences of an accident previously evaluated.

Based on the above discussion, it is concluded that this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendment does not change the plant configuration in a way which introduces a new potential hazard to the plant. Since design requirements continue to be met and the integrity of the reactor coolant system pressure boundary is not challenged, no new failure mode has been created. As a result, an accident which is different than already evaluated in the Updated Final Safety Analysis Report will not be created due to this change.

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

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

The margin of safety is not significantly reduced by this proposed change. The trip setpoint for the containment pressure intermediate-high-high function remains unchanged. With one channel inoperable, the remaining two channels will continue to initiate the protective function on a two-out-of-two logic. The action statement limits this condition to 6 hours after which time the inoperable channel must be placed in the trip condition. This action restores the function to be able to meet single failure criteria on a one-out-of-two logic basis.

The proposed revision to the charging pump discharge pressure will not change the flow limit on seal injection. The specification will continue to ensure that seal injection flow is limited. This will ensure that sufficient flow to the reactor core is provided during accident conditions.

The proposed changes to the Bases for seal injection flow are editorial in nature and do not affect the margin of safety.

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

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

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

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

NRC Project Director: John F. Stolz

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

Date of amendment request: August 29, 1996

Description of amendment request: The proposed amendment would revise the technical specifications (TSs) to reflect the elimination of T-factor adjustments in the Average Power

Range Monitors (APRM) setpoints, a decrease in the calibration frequency of the Local Power Range Monitors (LPMR), and an improvement in the calculation of Reactivity Anomaly.

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

This change replaces the APRM setpoints T-factor limit with power and flow-dependent minimum critical power ratio (MCPR) and linear heat generation rate (LHGR) limits. These new power and flow-dependent thermal limits eliminate the need for manual setpoint adjustment resulting from power peaking conditions. The new power and flow-dependent thermal limits are automatically applied by computer software during the calculation of the core thermal limits and, therefore, do not require manual setpoint adjustments based on the power peaking conditions in the reactor. Extensive transient analyses at a variety of power and flow conditions have been performed and were utilized to study the trend of transient severity without the setpoints T-factor limit. A large data base was established by analyzing limiting transients over a range of power and flow conditions. The data base included evaluations representative of a variety of plant configurations and parameters such that the conclusions drawn from the studies would be applicable to the broad range of boiling water reactors (BWRs). This data base was utilized to develop plant specific operating limits (MCPR and LHGR), which assures that margins to fuel safety limits are equal to or larger than those currently in existence with the APRM setpoints T-factor limit applied. Therefore, this change does not involve an increase in the probability of any event previously evaluated.

The consequences of an accident previously evaluated have not been increased because, in all cases, the new power and flow-dependent thermal limits (MCPR and LHGR) assure that margins to fuel safety limits are equal to or larger than those currently in existence with the APRM setpoints T-factor limit applied. Protection of other thermal limits for all previously analyzed events is accomplished by specific limits that are independent of the APRM setpoints T-factor. These are the power and flow-dependent MCPR Operating Limits which provide protection from fuel dryout and the rated maximum average planner linear heat generation rate (MAPLHGR) limit which provides protection of the peak clad temperature for the design basis accident-loss of coolant accident (DBA LOCA). Therefore, the proposed change does not involve a significant increase in the consequences of any event previously evaluated.

No new equipment is introduced by the change in the local power range monitor (LPRM) calibration frequency and, therefore, the probability for an accident previously evaluated is unchanged. The consequences of an accident can be affected by the thermal limits prior to the accident but LPRM chamber and cycle exposure have no significant effect on the calculated thermal limits. The thermal limit calculation is not significantly effected because the LPRM sensitivity versus exposure function is well defined. This allows accurate LPRM end-of-life calculations so that detectors can be replaced before their behavior significantly deteriorates. In the event deterioration is noted late in the cycle for a few chambers, they can be bypassed with no significant effect on uncertainties. Also, the total nodal power uncertainty remains less than the uncertainty assumed in the General Electric BWR Thermal Analysis Basis (GETAB) safety limit. Therefore, the thermal limit calculation is not affected by the LPRM calibration frequency and the consequences of an accident previously evaluated are not changed.

The change in the parameters used to measure reactivity for calculation of the reactivity anomaly has no affect on either the consequences or the probability of an accident previously evaluated because the allowed reactivity anomaly criteria is unchanged. The only change is the parameters used to measure reactivity.

Therefore, the proposed elimination of the APRM setpoints T-factor maintains adequate off-rated MCPR and LHGR margin for all operating conditions. Also, the change in the LPRM calibration frequency continues to maintain the accuracy of the thermal limit calculation. Therefore, the consequences of an accident previously evaluated are not affected by this change. Finally, the change in the parameters used to measure reactivity for calculation of the reactivity anomaly has no affect on either the consequences nor the probability of an accident previously evaluated. Since no new plant equipment is introduced by any of the proposed changes, the probability of accidents previously evaluated are not changed. Therefore, none of the proposed changes involve an increase in the probability or consequences of any event previously evaluated.

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

This change only replaces the APRM setpoints T-factor limit with power and flow-dependent MCPR and LHGR limits, changes the LPRM calibration frequency, and a change to the parameter(s) used to measure reactivity. None of the proposed changes involve any new modes of operation or any plant modifications. Therefore, the proposed changes do not create the possibility of a new or different type of accident from any accident previously analyzed.

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

The replacement of the APRM setpoints T-factor limit with power and flow-dependent thermal limits has been confirmed to provide adequate MCPR and LHGR protection at all reactor operation conditions. Operation with higher peaking without APRM gains or flow bias trip setpoints adjustment does not involve a reduction in a margin of safety because the higher power peaking resulting from elimination of the APRM setpoints T-factor has been analyzed to assure that the margins to fuel safety limits are equal to or larger than those currently in existence with the APRM setpoints T-factor limit applied. Therefore, the replacement of the APRM setpoint T-factor with power and flow-dependent thermal limits does not involve a reduction in the margin of safety.

Protection of other thermal limits for all previously analyzed events is accomplished by specific limits that are independent of the APRM setpoint T-factor limit. These are the power and flow-dependent

MCPR Operating Limits which provide protection from fuel dryout and the rated MAPLHGR limit which provides protection of the peak clad temperature for the DBA LOCA.

The margin of safety can be affected by the thermal limits prior to an accident but LPRM chamber exposure and cycle exposure have no significant effect on the calculated thermal limits. The thermal limit calculation is not significantly affected because the LPRM sensitivity versus exposure function is well defined. This allows accurate LPRM end of life calculations so that detectors can be replaced before their behavior significantly deteriorates. In the event deterioration is noted late in the cycle for a few chambers, they can be bypassed with no significant effect on uncertainties. Also, the total nodal power uncertainty remains less than the uncertainty assumed in the GETAB safety limit. Therefore neither the thermal limit calculation nor the margin of safety are affected by the LPRM calibration.

The change in the parameters used to measure reactivity for calculation of the reactivity anomaly has no affect on the margin of safety because the allowed reactivity anomaly criteria is unchanged. The only change is the parameters used to measure reactivity.

Neither the change to APRM setpoints T-factor nor the change to the LPRM calibration frequency significantly effects the thermal limits calculation, and, therefore, do not result in an increase in core damage frequency. The change in the parameters used to measure reactivity for calculation of the reactivity anomaly has no affect on the core damage frequency because the allowable reactivity anomaly criteria remains unchanged. Therefore, 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 request involves no significant hazards consideration.

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

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005

NRC Project Director: William D. Beckner

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

Date of amendment request: August 29, 1996

Description of amendment request: The proposed amendment would provide a revision to the reactor pressure vessel (RPV) surveillance capsule withdrawal schedule for the River Bend Station. The first surveillance capsule would be withdrawn at 10.4 effective full power years (EFPY) rather than at 6EFPY.

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.

Pressure-temperature (P-T) limits (RBS Technical Specifications Figure 3.4.11-1) are imposed on the reactor coolant system to ensure that adequate safety margins against nonductile or rapidly propagating failure exist during normal operation, anticipated operational occurrences, and system hydrostatic tests. The P-T limits are related to the nil-ductility reference temperature, RT_{NDT}, as described in ASME Section III, Appendix G. Changes in the fracture toughness properties of RPV beltline materials, resulting from the neutron irradiation and the thermal environment, are monitored by a surveillance program in compliance with the requirements of 10CFR50, Appendix H. The effect of neutron fluence on the shift in the nil-ductility reference temperature of pressure vessel steel is predicted by methods give in Regulatory Guide 1.99, Rev. 2.

River Bend's current P-T limits were established based on adjusted reference temperatures developed in accordance with the procedures prescribed in Reg. Guide 1.99, Rev. 2, Regulatory Position 1. Calculation of adjusted reference temperature by these procedures includes a margin term to ensure conservative, upper-bound values are used for the calculation of the P-T limits. Revision of the first capsule withdrawal schedule will not affect the P-T limits because they will continue to be established in accordance with Regulatory Position 1 (or other NRC-approved) procedures. When permitted (two or more credible surveillance data sets available), Regulatory Position 2 (or other NRC-approved) methods for determining adjusted reference temperature will be followed.

This change is not related to any accidents previously evaluated. The proposed change is a revision of the Withdrawal Time for the first surveillance capsule as given in Technical Requirements (TR) Table 3.4.11-1 from 6 EFPY to 10.4 EFPY. This change will not affect P-T limits as given in RBS Technical Specifications Figure 3.4.11-1 or USAR Figures 5.3-4a and 5.3-4b. This change will not affect any plant safety limits or limiting conditions of operation. The proposed change will not affect reactor pressure vessel performance as no physical changes are involved and RBS vessel P-T limits will remain conservative in accordance with Reg. Guide 1.99, Rev. 2 requirements. The proposed change will not cause the reactor pressure vessel or interfacing systems to be operated outside of their design or testing limits. Also, the proposed change will not alter any assumptions previously made in evaluating the radiological consequences of accidents. Therefore, the probability or consequences of accidents previously evaluated will not be increased by the proposed change.

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

The proposed change is a revision of the Withdrawal Time in TR Table 3.4.11 for the first RPV material surveillance capsule from 6 EFPY to 10.4 EFPY. This proposed change does not involve a modification of the design of plant structures, systems, or components. The proposed change will not impact the manner in which the plant is operated as plant operating and testing procedures will not be affected by the change. The proposed change will not degrade the reliability of structures, systems or components important to safety (ITS) as equipment protection features will not be deleted or modified, equipment redundancy or independence will not be reduced, supporting system performance will not be downgraded, the frequency of operation of ITS equipment will not be increased, and increased or more severe testing of ITS equipment will not be imposed. No new accident types or failure modes will be introduced as a result of the proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from that previously evaluated.

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

As stated in the River Bend SER, ''Appendices G and H of 10CFR50 describe the conditions that require pressure-temperature limits and provide the general bases for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins at least as great as those recommended in the ASME Code, Section III, Appendix G. .... Until the results from the reactor vessel surveillance program become available, the staff will use RG 1.99, Revision 1 [now Revision 2] to predict the amount of neutron irradiation damage. ... The use of operating limits based on these criteria--as defined by applicable regulations, codes, and standards--will provide reasonable assurance that nonductile or rapidly propagating failure will not occur, and will constitute an acceptable basis for satisfying the applicable requirements of GDC 31.''

Bases for RBS Technical Specification 3/4/11 states: ''The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB [Reactor Coolant Pressure Boundary], a condition that is unanalyzed. ... Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.''

The proposed change will not affect any safety limits, limiting safety system settings, or limiting conditions of operation. The proposed change does not represent a change in initial conditions, or in a system response time, or in any other parameter affecting the course of an accident analysis supporting the Bases of any Technical Specification. The proposed change does not involve revision of the P-T limits but rather a revision of the Withdrawal Time for the first surveillance capsule. The current P-T limits were established based on adjusted reference temperatures for vessel beltline materials calculated in accordance with Regulatory Position 1 of Reg. Guide 1.99, Rev. 2. P-T limits will continue to be revised as necessary for changes in adjusted reference temperature due to changes in fluence according to Regulatory Position 1 until two or more credible surveillance data sets become available. When two or more credible surveillance data sets become available, P-T limits will be revised as prescribed by Regulatory Position 2 of Reg. Guide 1.99, Rev. 2 or other NRC-approved guidance. Therefore, the proposed changes do not involve a significant reduction 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: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005

NRC Project Director: William D. Beckner

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 23, 1996

Description of amendment request: The proposed amendment would revise the Crystal River Unit 3 (CR 3) technical specifications (TS) to delete a note associated with Surveillance Requirement (SR) 3.3.7.1 for the Engineered Safeguard Actuation System (ESAS) Automatic Actuation Logic. Applicable TS Bases will also be revised to reflect the proposed TS change.

SR 3.3.7.1 requires periodic testing of the ESAS automatic actuation logic matrix to demonstrate that the required logic combinations are operable. When the ESAS automatic actuation logic is placed in an inoperable status solely for performing of this surveillance, the note associated with the SR 3.3.7.1 provides relief in that it allows not entering into applicable Conditions and Required Actions for up to 8 hours, provided the associated engineering safeguards (ES) function is maintained. The licensee has determined that because of the CR 3 design of the ESAS System and the way the test is performed, maintenance of the ''associated ES function'' is not possible. Thus, the note does not provide the relief intended and therefore, the licensee proposes to delete the note. During the performance of the ESAS test and bypassing the associated ES function, the licensee proposes to enter into applicable TS Conditions.

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

1. The proposed change will not significantly increase the probability or consequences of an accident previously evaluated because unavailability of equipment is recognized in the design of the plant and in the Technical Specifications. The probability and consequences of accidents previously evaluated are bounded by the evaluations done for the allowed outage time of the associated functions.

2. The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated because the bypassing of ES functions for testing purposes does not place the plant in a configuration which would allow the possibility of a new or different kind or accident to be created.

3. The proposed change will not involve a significant reduction to the margin of safety because deleting the NOTE does not effect the way the test is performed. The test is required by the Technical Specifications and will still be performed in the same manner. Thus, there is no change in the unavailability of the system as a result of this change and the margin of safety is not reduced.

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

Local Public Document Room location: 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: Frederick J. Hebdon

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 27, 1996

Description of amendment request: The proposed amendment would revise the Crystal River 3 (CR3) post-accident monitoring (PAM) instrumentation technical specification (TS). Specifically, the following TS changes are proposed:

A. Table 3.3.17-1, Function 8: The descriptor is changed from ''Containment Pressure (Narrow Range)'' to ''Containment Pressure (Expected Post-Accident Range).''

B. Table 3.3.17-1, Function 18: The required channels for Core Exit Temperature (Backup) is changed from ''2 sets of 5'' to ''3 per core quadrant.''

C. Table 3.3.17-1: A new Function 20 is added and designated as ''Low Pressure Injection Flow.''

D. Table 3.3.17-1: A new Function 21 is added and designated as ''Degrees of Subcooling.''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration (the letters A, B, C and D correspond to the proposed TS changes), which is presented below:

1. The proposed changes will not significantly increase the probability or consequences of an accident previously evaluated because:

A/B. The changes in containment pressure and core exit thermocouple nomenclature do not reflect any physical changes to the facility.

C/D.The addition of low pressure injection flow and degrees of subcooling to the Post-Accident Monitoring Instrumentation LCO is being done to comply with a commitment made during the technical specification improvement program to include in the technical specifications, that instrumentation which monitors variables classified as Type A in accordance with Regulatory Guide 1.97. These two variables have recently been re-classified as Type A. The associated instruments are used after an accident occurs to prompt the operators to take certain mitigative actions. Therefore, the probability of an accident occurring is unaffected. As part of the re-classification of these variables to Type A, the associated monitoring instrumentation will be under more strict surveillance and control, which provides additional assurance that the prescribed manual operator actions will be implemented when necessary. This, in turn, assures the previously evaluated accident consequences remain valid.

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

A/B. The changes in containment pressure and core exit thermocouple nomenclature do not reflect any physical changes to the facility. The changes provide clarification for the instruments which are required to comply with the LCO.

C/D. The addition of low pressure injection flow and degrees of subcooling to the Post-Accident Monitoring instrumentation LCO is being done to comply with a commitment made during the technical specification improvement program to include in the technical specifications, that instrumentation which monitors variables classified as Type A in accordance with Regulatory Guide 1.97. These two variables have been re-classified as Type A. The associated instruments are used after an accident occurs to prompt the operators to take certain mitigative actions. Since the instrumentation is used only post-accident, these changes do not create the possibility of a new or different kind of accident.

3. The proposed change will not involve a significant reduction to the margin of safety because:

A/B. The changes in containment pressure and core exit thermocouple nomenclature have no affect on the margin of safety. The changes provide clarification of the technical specifications. This reduces the potential for confusion regarding this instrumentation.

C/D. The addition of low pressure injection flow and degrees of subcooling to the post-accident monitoring instrumentation table adds controls on the OPERABILITY of post-accident monitoring instrumentation providing greater assurance it will be available should an accident occur.

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: Frederick J. Hebdon

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

Date of amendment request: September 5, 1996

Description of amendment request: The proposed change deletes License Condition 2.C.5, Integrated Implementation Schedule.

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

In accordance with 10CFR50.92, NNECO has reviewed the attached proposed change and has concluded that it does not involve a significant hazards consideration (SHC). The basis for this is that the three criteria of 10CFR50.92(c) are not compromised. The proposed change does not involve an SHC because the change would not:

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

Operation of the facility in accordance with the proposed change would result in a change in an administrative process for prioritizing and scheduling projects and engineering evaluations. With the limited number of NRC required projects remaining to be implemented, the IIS [Integrated Implementation Schedule] is no longer required to schedule resources for the remaining topics. Since this license condition only involves an administrative process, it does not directly affect the design or operation of the plant. Therefore, no accident analyses are affected by the change, and the change does not increase the probability or consequences of any previously evaluated accident.

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

The proposed license modification removes a requirement relating to the scheduling of modifications and engineering evaluations. Because the license condition addresses only an administrative scheduling mechanism, it does not affect directly the design or operation of the plant. Therefore, the proposed change does not create a different kind of accident from those previously analyzed.

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

The proposed license modification removes a requirement relating to the scheduling of modifications and engineering evaluations. The original purpose of the IIS and the ISAP [Integrated Safety Assessment Program] was to prioritize and schedule modifications and engineering evaluations in a manner that was agreed upon by both NNECO and the NRC. These programs were especially important to Millstone Unit No. 1 for priorization of topics associated with the SEP [Systematic Evaluation Program] and the TMI [Three Mile Island] Action Plan. This program is considered to be no longer necessary. Modifications and engineering evaluations will be scheduled and prioritized using other methodologies. Since this change involves an administrative process only, there is no direct impact on the design or operation of the plant, and therefore, no significant reduction in the margin of safety.

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

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

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

NRC Project Director: Phillip F. McKee

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

Date of application for amendments: August 27, 1996

Description of amendment request: The proposed amendment revises the required value of control rod drive (CRD) system pressure in technical specification (TS) 3.10.8, ''Shutdown Margin (SDM) Test-Refueling.''

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 any accident previously evaluated.

The proposed changes are purely administrative and do not involve any physical changes to plant SSC [systems, structures and components]. The change in the minimum CRD charging water header pressure from 955 psig to 940 psig was previously approved in TS Amendments Nos. 211 and 216 for PBAPS [Peach Bottom Atomic Power Station], Units 2 and 3. TS Change Request 95-12 was incomplete by inadvertently failing to identify the need to change requirement (f) of LCO [Limiting Condition for Operation] 3.10.8. Therefore, the proposed changes will not increase the probability of occurrence or the consequences of an accident previously evaluated in the SAR [safety analysis report].

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 are purely administrative and do not involve any physical changes to plant SSC. The proposed changes do not allow plant operation in any mode that is not already evaluated in the SAR. Therefore, the possibility of a different type of accident than previously evaluated in the SAR is not created.

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

The proposed changes are purely administrative and have no impact on any safety analysis assumptions or margins of safety. A change to SR 3.10.8.6 was approved by the NRC by TS Amendment Nos. 211 and 216. LCO 3.10.8 requirement (f) should have been changed at the same time to reflect a minimum CRD charging water pressure of 940 psig. Changing LCO 3.10.8 requirement (f) to reflect TS Amendment Nos. 211 and 216 is purely administrative, and therefore, 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: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105

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

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: May 20, 1996

Description of amendment request: The proposed Technical Specifications (TS) changes would revise TS Sections 3/4.4.9.2, 3/4.9.11.1, 3/4.9.11.2, and the associated TS Bases 3/4.4.9 and 3/4.9.11, to more clearly describe that the Residual Heat Removal (RHR) system Shutdown Cooling mode of operation consists of four (4) ''subsystems.'' These TS sections pertain to plant operations during Operational Conditions (OPCONs) 4, ''Cold Shutdown'' and 5, ''Refueling.'' In addition, the proposed TS change would make administrative changes to TS Section 3/4.4.9.1 to ensure consistency in terminology regarding the description of Shutdown Cooling ''subsystems.'' The proposed TS changes are consistent with the guidance delineated in the Improved TS (i.e., NUREG-1433, Revision 1, ''Standard Technical Specifications General Electric Plants, BWR/4,'' dated April 1995) which indicates that the RHR Shutdown Cooling mode of operation is comprised of two (2) loops and four (4) subsystems (i.e., two (2) subsystems per loop).

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 Technical Specifications (TS) changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS changes do not involve any physical changes to plant structures systems, or components. The RHR [Residual Heat Removal] Shutdown Cooling mode of operation is manually controlled and is not required for accident mitigation. The RHR system will continue to function as designed in all modes of operation. The consequences of equipment malfunction are not changed from those in existing analyses, with no increase in onsite or offsite radiological effects. The RHR system will continue to function as designed to mitigate the consequences of an accident and resultant onsite and offsite radiological effects remain as previously evaluated. The proposed TS changes will revise the TS to more clearly describe the RHR system configuration in OPCONs 4 and 5. The proposed changes are consistent with the guidance stipulated in NUREG-1433, Revision 1.

The four (4) ''subsystem'' Shutdown Cooling designation permits operability of only one (1) RHR heat exchanger for Shutdown Cooling service in Operational Conditions (OPCONs) 4 and 5, as long as both associated RHR pumps are operable and alignable for Shutdown Cooling. TS requirements for RHR Shutdown Cooling operation in Hot Shutdown, Suppression Pool Spray, and Suppression Pool Cooling continue to require two (2) independent loops to be operable in OPCONs 1, 2, and 3*, meaning both RHR heat exchangers will still be required to be operable throughout OPCON 3.

The four (4) ''subsystem'' Shutdown Cooling designation has no effect on the required operability of the Residual Heat Removal Service Water (RHRSW) system. As required by TS Section 3.7.1.1, the RHRSW subsystem(s) associated with the required operable RHR heat exchanger(s) will continue to remain operable. Each operable RHRSW subsystem consists of two (2) operable pumps and the required operable flowpath to provide decay heat removal via the associated RHR heat exchanger.

The RHRSW system piping is designed, fabricated, inspected, and tested in accordance with the requirements of ASME [American Society of Mechanical Engineers], Section III Class 3, and each RHRSW subsystem is single active failure proof in that the failure of a motor-operated valve, diesel generator, or pump does not prevent the system from performing its safety function.

The required availability of four (4) loops of the Low Pressure Coolant Injection (LPCI) mode of RHR during OPCONs 1, 2, and 3 as required by TS Section 3.5.1 is not impacted by the four (4) ''subsystem'' Shutdown Cooling designation. No change to any RHR system instrumentation logic, required Emergency Core Cooling System (ECCS) availability, or method of operation is involved.

NUREG-1433, Revision 1, also re-affirms that each Shutdown Cooling ''subsystem'' is considered operable if it can be manually aligned, remotely or locally, in the shutdown cooling mode for removal of decay heat. Thus, a LPCI-dedicated pump can be aligned for LPCI automatic initiation, yet still be considered part of an operable shutdown cooling subsystem as long as it can be re-aligned for Shutdown Cooling.

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 involve any physical changes to plant structures, systems, or components. The RHR system will continue to function as designed in all modes of operation. No new accident type is created as a result of the proposed changes. No new failure mode for any equipment is created. The changes are consistent with the guidance provided in NUREG-1433, Revision 1, pertaining to RHR Shutdown Cooling operation in OPCONs 4 and 5.

The four (4) ''subsystem'' Shutdown Cooling designation has no effect on the required operability of the RHRSW system. The RHRSW subsystem(s) associated with the required operable RHR heat exchanger(s) will continue to remain operable as required by TS Section 3.7.1.1. Each operable RHRSW subsystem consists of two (2) operable pumps and the required operable flowpath to provide decay heat removal via the associated RHR heat exchanger.

The RHRSW system piping is designed, fabricated, inspected, and tested in accordance with the requirements of ASME, Section III, Class 3, and each RHRSW subsystem is single active failure proof in that the failure of a motor-operated valve, diesel generator, or pump does not prevent the system from performing its safety function.

The required availability of four (4) loops of the LPCI mode of RHR during OPCONs 1, 2, and 3 as required by TS Section 3.5.1 and 3.5.2 is not impacted by the four (4) ''subsystem'' Shutdown Cooling designation. No change to any RHR system instrumentation logic, required ECCS availability, or method of operation is involved.

NUREG-1433, Revision 1, also re-affirms that each Shutdown Cooling ''subsystem'' is considered operable if it can be manually aligned, remotely or locally, in the Shutdown Cooling mode for removal of decay heat. Thus, a LPCI-dedicated pump can aligned be [sic] [be aligned] for automatic LPCI initiation, yet still be considered part of an operable shutdown cooling subsystem as long as it can be re-aligned for Shutdown Cooling.

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

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

Although the Bases for TS Sections 3/4.4.9.2, 3/4.9.11.1, and 3/4.9.11.2 are being revised in support of this proposed TS change, the changes only involve providing clarification regarding the designation of the RHR Shutdown Cooling operation configuration in OPCONs 4 and 5. The proposed TS changes do not involve any physical changes to plant structures, systems, or components. The RHR system will continue to function as designed in all modes of operation. The consequences of equipment malfunction are not changed from those in existing analyses, with no increase in onsite or offsite radiological effects. The RHR system will continue to function as designed to mitigate the consequences of an accident and resultant onsite and offsite radiological effects remain as previously evaluated. The proposed changes are consistent with the guidance stipulated in NUREG-1433, Revision 1.

The four (4) ''subsystem'' Shutdown Cooling designation has no effect on the required operability of the RHRSW system. As required by TS 3.7.1.1, the RHRSW subsystem(s) associated with the required operable RHR heat exchanger(s) will continue to remain operable. Each operable RHRSW subsystem consists of two (2) operable pumps and the required operable flowpath to provide decay heat removal via the associated RHR heat exchanger.

The RHRSW system piping is designed, fabricated, inspected, and tested in accordance with the requirements of ASME, Section III, Class 3, and each RHRSW subsystem is single active failure proof in that the failure of a motor-operated valve, diesel generator, or pump does not prevent the system from performing its safety function. (In the same manner that manual action may be required for RHR system alignment in OPCONs 4 and 5 with one (1) RHR heat exchanger operable, a failure of the motor-operated RHRSW inlet or outlet heat exchanger isolation valves may require manual positioning for the required alignment.)

The required availability of four (4) loops of the LPCI mode of RHR during OPCONs 1, 2, and 3* as required by TS Section 3.5.1 is not affected by the four (4) ''subsystem'' Shutdown Cooling configuration. No change to any RHR system instrumentation logic, required ECCS availability, or method of operation is involved.

NUREG-1433, Revision 1, also re-affirms that each Shutdown Cooling ''subsystem'' is considered operable if it can be manually aligned, remotely or locally, in the Shutdown Cooling mode for removal of decay heat. Thus, a LPCI-dedicated pump can be aligned for LPCI automatic initiation, yet still be considered part of an operable Shutdown Cooling ''subsystem'' as long as it can be re-aligned for Shutdown Cooling.

Therefore, the proposed TS 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: Pottstown Public Library, 500 High Street, Pottstown, PA 19464

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

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: June 28, 1996

Description of amendment request: The proposed Technical Specifications (TS) changes would incorporate performance-based testing, in accordance with 10 CFR Part 50, Appendix J, ''Primary Reactor Containment Leakage Testing For Water-Cooled Power Reactors,'' Option B. This option allows utilities to extend the frequencies of the Type A Containment (ILRT) Leak Rate Test and Type B and C Local Leak Rate Tests (LLRTs) based on the performance and design of the containment and components.

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

Incorporation of the new 10 CFR 50, Appendix J, Option B at LGS, Units 1 and 2 does not increase the probability of occurrence of an accident previously evaluated. The containment structure including its isolation capability is not an accident initiator.

These changes do not involve any changes to the containment structure, system or components which could increase the probability of occurrence of an accident previously evaluated or act as a new accident initiator. Implementation of the proposed changes will affect the manner in which these structures, systems, or components (SSCs) are tested; however, the new testing schedule is not an initiator of any analyzed event. No equipment changes are involved with adoption of Option B; therefore, performance-based test intervals for Type A, B, and C tests do not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated. No physical changes are being made to the plant, nor are there any changes being made in the operation of the plant as the result of increasing the test intervals. Additionally, the proposed TS changes will not alter the operation of equipment available for the mitigation of accidents or transients, therefore, this change will not result in any significant increase to onsite or offsite dose previously evaluated. The potential for time-based and activity-based failure mechanisms which could lead to excessive containment leakage has been determined to be minimal. Performance-based test intervals for Type A, B, and C tests will not alter any safety limits which ensure the integrity of fuel barriers, and will not increase the primary containment leakage limits.

Performance-based test intervals for Type A, B, and C leak tests do not increase the consequences of an accident previously evaluated. NUREG-1493 concluded that reducing the frequency of Type A tests from the current three per ten years to one per ten years was found to lead to an imperceptible increase in risk. NUREG-1493 includes the results of a sensitivity study performed to explore the risk impact of several alternative leak rate test schedules. The estimated increase in population exposure risk ranged from 0.02% to 0.14%. The risk impact was determined to be very small since Type B and C testing (local leak rate tests) detect a very large percentage of overall containment leakages. The percentage of leakages detected by Type A tests is very small. Past test results experienced at Limerick Units 1 and 2 concur with these determinations. NUREG-1493 also concluded that the overall unit risk is not very sensitive to changes in containment leakage rates. Given the insensitivity of risk to containment leak rates and the small fraction of leak paths detected solely by the Type A tests, increasing the interval between Type A tests is possible with minimal impact on public risk.

NUREG-1493 also concluded that, based on a model of component failure with time, the performance-based alternatives to current, local-leakage testing requirements are feasible without significant risk impact. The LGS design and past performance is bounded by the NUREG study. The NUREG model indicated that the number of components tested could be reduced by about 60% with less than a three-fold increase in the incremental risk due to containment leakage. Since under existing requirements, leakage contributes less than 0.1 percent of overall accident risk, the overall impact is very small.

Therefore, the proposed TS changes will not involve a significant 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.

Performance-based test intervals for Type A, B, and C leak tests do not introduce a new or different type of accident or create the possibility of a different type of malfunction of equipment important to safety than previously evaluated. No physical changes are being made to the plant, nor are there any changes being made in the operation of the plant as the result of increasing the test intervals. No new failure modes of plant equipment previously evaluated will be introduced. Additionally, the TS changes will not alter the operation of equipment available for the mitigation of accidents or transients. The safety function of the primary containment will be retained since the containment will continue to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents previously evaluated.

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

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

The margin of safety is not reduced as a result of adopting 10 CFR 50, Appendix J, Option B. The effect of increasing containment leakage rate testing intervals was evaluated in NUREG-1493 using historical industry leakage rate testing results. Performance history at LGS is consistent with the conclusions reached in NUREG-1493 and NEI 94-01. The results of the NUREG evaluation conclude that the increased safety risk corresponding to the extended test intervals is small (less than 0.1% of total risk). The revised TS will continue to maintain the allowable leakage rate for the Type A tests. In addition, the requirement to perform a periodic general visual inspection of the primary containment has been maintained at the original interval of three times in 10 years as part of the performance-based leakage rate testing program.

The risk of a non-detectable increase of primary containment leakage is considered to be negligible due to the conclusion that 10 CFR 50, Appendix J, Type B and C testing program will continue to be conducted between Type A tests. A review of previous LGS Type A test results has concluded that the only failure mechanisms are activity-based. There is no indication of time-based failures that would not be identified during the performance of Type B and C tests. Therefore, we have concluded that the proposed adoption of the Option B intervals would not result in a non-detectable primary containment leakage rate in excess of the allowable value (i.e., 0.5% wt/day) established by the LGS TS.

The proposed TS will continue to maintain the allowable leakage rate for the combined Type B and C tests. As supported by the findings of NUREG-1493, the percentage of leakages detected by Type A tests is small (as stated above) and Type B and C leakage tests are capable of detecting more than 97% of containment leakages and virtually all such leakages are identified by local leak rate tests of containment isolation valves. The Type B and C test intervals will be established through the PCLRTP for each component based on design and previous LGS test performance history.

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

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

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

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

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 25, 1996

Description of amendment request: The amendments would relocate to the Salem Updated Final Safety Analysis Report the list of containment isolation valves that are currently located in Table 3.6-1 of Technical Specification 3.6.3. In addition, references to the table in specifications 1.7, 3.6.1, and 3.6.3 are being updated.

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

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

The proposed changes simplify the TS, meet the regulatory requirements for control of containment isolation, and are consistent with the guidance provided in Generic Letter (GL) 91-08, ''Removal of Component Lists from Technical Specifications.'' The procedural details of TS Table 3.6-1 have not been changed, only relocated to a different controlling document, the Salem Update [sic] [Updated] Final Safety Analysis Report (UFSAR). The proposed changes are administrative in nature, should result in improved administrative practices, and do not affect plant operations.

The probability of occurrence of a previously evaluated accident is not increased because this change does not introduce any new potential accident initiating conditions. The consequences of an accident previously evaluated is not increased because the ability of containment to restrict the release of any fission product radioactivity to the environment will not be degraded by this change.

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

The proposed changes are administrative in nature, do not result in a physical alterations or changes to the operation of the plant, and cause no change in the method by which any safety-related system performs its functions. Therefore, this proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The administrative change to relocate TS Table 3.6-1 to the UFSAR does not alter the basic regulatory requirements for containment isolation and will not adversely affect the containment isolation capability for credible accident scenarios. Adequate control of the content of the relocated table is assured by the 10CFR50.59 review process.

The proposed relocation of TS Table 3.6-1 does not alter the requirements for CIV operability currently in the TS. the Limiting Condition for Operation and the Surveillance Requirements would be retained in the revised TS. Therefore, the proposed changes will not affect the meaning, application, and function of the current TS requirements for the CIVs.

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

Local Public Document Room location: Salem Free Public library, 112 West Broadway, Salem, NJ 08079

Attorney for licensee: 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 25, 1996

Description of amendment request: The amendments would change Technical Specification 3/4.8.1, ''Electrical Power Systems,'' to revise the Emergency Diesel Generator (EDG) voltage and frequency limits as a result of updated EDG load calculations and to eliminate ambiguity in the testing methodology for EDG start timing.

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

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

Since no change is being made to the offsite power supplies, or to any system or component that interfaces with the offsite power supplies, there is no change in the probability of a Loss of Offsite Power Accident.

The proposed changes provide the necessary conservatism for voltage and frequency to ensure the EDGs are not run in an overloaded condition and that driven equipment is not damaged during steady state operation following a Loss of Offsite Power coincident with a Loss of Coolant Accident. Since the narrower band of voltage and frequency for the isochronous mode continues to ensure proper steady state operation of the EDG and associated driven equipment, there is no change in the consequences of an accident previously evaluated.

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

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

The proposed amendment does not result in any design or physical configuration changes to the EDGs. Proposed changes made to the testing parameters and testing methodology will not cause a new or different accident since the EDGs are used for accident mitigation and no new failure modes are being introduced. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed amendment provides further conservatism to the voltage and frequency band currently specified in the TSs. The proposed voltage and frequency changes ensure the EDG will not be overloaded from an over-frequency condition and driven equipment will not be damaged from an over-voltage condition.

The control system is set to control the EDG voltage within the bands specified in the requested changes. The changes are consistent with current calculations and within the capability of the controls. Since the narrower band of voltage and frequency for the isochronous mode is bounded by the existing TS, there is no change in the margin of safety. The increased band for droop mode will ensure the EDG is capable of operating in accordance with normal offsite power parameters and does not reduce the margin of safety.

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

Local Public Document Room location: Salem Free Public library, 112 West Broadway, Salem, NJ 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: October 1, 1996

Description of amendment request: The proposed amendments would change Technical Specifications (TSs) 3/4.7.1.5, ''Main Steam Line Isolation Valves (MSIVs),'' and 3/4.3.2, ''Engineered Safety Feature Actuation System Instrumentation.'' These changes are needed to accommodate entry into Modes 3 and 2 prior to performing MSIV closure time testing in Mode 2. The proposed amendments would also allow for the repair and testing of inoperable MSIVs in certain operating Modes, and would change the low steam line pressure trip setpoint value for safety injection to make it consistent with the previously approved value for steam line isolation.

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

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

The isolation capability of the MSIVs and the protective functions of the low steam line pressure channels are necessary for accident mitigation and do not impact the probability of an accident. MSIV testing in the higher modes is necessary to obtain conditions which enable testing of the MSIVs. These conditions are consistent with the current accident analyses for main steam line breaks and secondary system depressurization. Failure of a MSIV, which could be encountered during testing, is accounted for in the accident analyses.

Provisions for entering Mode 2 within six hours with an inoperable MSIV allows operators to remove the plant from power generation in a more controlled manner without challenging plant safety systems and is consistent with other plant shutdown TS (i.e., TS 3.0.3). The additional six hours to Hot Shutdown, should MSIV closure be infeasible, does not result in a significant increase in the probability or consequence of an accident since this is a very small incremental time addition. The values for the low steam line pressure safety injection are higher and are bounded by the present accident analysis. The elimination of the obsolete stroke time of eight seconds is editorial in nature. As a result, the changes proposed do not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

The proposed changes do not involve any modifications to existing plant equipment, do not alter the function of any plant systems, do not introduce any new operating configurations or new modes of plant operation, nor change the safety analyses. The proposed changes will, therefore, not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

MSIV testing in Mode 2 is within the currently analyzed plant operation as discussed in the Updated Final Safety Analysis Report (UFSAR) Sections 10.3 and 15.4. These UFSAR sections address performance of the TS surveillance test at or near 1000 psig Steam Generator pressure to assure main steam isolation occurs within the accident conditions, where Steam Generator pressure may be lower during Mode 1 operation. The test methodology demonstrating MSIV operability is consistent with the accident analysis.

Operation in Modes 2 and 3 with one or more isolation valve inoperable and in the closed position does not impact the margin of safety since the valves are already performing the safety function.

The protective functions that occur as a result of the low steam line pressure initiating signal remain bounded by the values assumed in the safety analyses. That is, the protective functions that occur as a result of this initiating signal already assume a setpoint that is conservative for the revised value. The change to the setpoint eliminates conflicting information in the TS.

Therefore, the proposed changes 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: Salem Free Public library, 112 West Broadway, Salem, NJ 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 No. 50-311, Salem Nuclear Generating Station, Unit No. 2, Salem County, New Jersey

Date of amendment request: September 20, 1996, as supplemented September 30, 1996

Description of amendment request: The proposed amendment would change Technical Specification 4.7.7.b.4 to indicate that the specified flowrate for the Auxiliary Building Exhaust Air Filtration System applies only to system testing.

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

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

The accident considered in this proposed change is the Loss of Coolant Accident (LOCA) as described in Section 15.4 of the UFSAR [Updated Final Safety Analysis Report]. The assumption is that: ''The Auxiliary Building Ventilation System will discharge the vapor (from recirculation liquid leakage) to the atmosphere through charcoal filters which have an efficiency of 90 percent.'' As such the system acts to limit the total offsite and control room radiation doses following a LOCA.

The Auxiliary Building Ventilation System [ABVS] is designed to maintain the Auxiliary Building at a negative pressure with respect to the atmosphere during normal and emergency operation. Filtration of radio-iodines is accomplished by administratively aligning the ECCS [emergency core cooling system] equipment areas exhaust flows to the standby charcoal adsorber bed if required. The ABVS has no direct impact on reactor operation or on any system connected to the Reactor Coolant Pressure Boundary.

The emergency operation of the Auxiliary Building Ventilation System is not affected by the proposed changes. The acceptance criteria for system performance are not modified by the requested change. The change clarifies the intent of SR [surveillance requirement] 4.7.7.b.4 and the basis for the flowrates used for system acceptance testing. It has been determined that operation of the system at lower flow rates than those specified for surveillance testing is conservative with respect to the radio-iodine removal efficiency assumed for the charcoal adsorber. A higher removal efficiency results in lower total exposures at the site boundary and within the control room. Additionally, the system is capable of maintaining the required negative pressure at the reduced flowrate.

Given the above, it is concluded that the proposed change does not result in an increase in the probability or consequences associated with previously analyzed accidents.

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

The proposed amendment does not result in any design or operational change to the ABVS, to the Nuclear Steam Supply System, to the ECCS System, to the Containment Building, to the fuel or to the electrical power supplies. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Specification 3/4.7.7 and the associated bases were reviewed to determine if the proposed changes result in a reduction in the margin of safety. The change to SR 4.7.7.b.4 continues to assure that the system is operated consistent with the assumptions of the accident analysis. The proposed changes to Bases 3/4.7.7 clarify the basis for flowrates associated with ABVS surveillance test requirements. All changes result in ABVS operation that is just as conservative as that assumed in existing analyses.

The proposed changes do not involve the addition or modification of plant equipment, are consistent with the design basis of the ABVS as described in the UFSAR, and appropriately limit operation to be consistent with the assumptions of the accident analysis. As such there is no 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: Salem Free Public library, 112 West Broadway, Salem, NJ 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

Previously Published Notices Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the Federal Register on the day and page cited. This notice does not extend the notice period of the original notice.

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

Date of application for amendments: June 17, 1996

Brief description of amendments request: The proposed amendments would modify the technical specifications to change (1) the reference method for calculating dose conversion factors (DCFs) to be used in dose calculations, and (2) the upper and lower limits for operating pressurizer pressure to account for new instrument uncertainties and to reduce the allowed operating band.

Date of individual notice in Federal Register: September 11, 1996 (61 FR 47963)

Expiration date of individual notice: October 11, 1996

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

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

Date of application for amendments: June 28, 1996

Brief description of amendments request: The proposed amendments would modify the technical specifications to increase the minimum required amount of anhydrous trisodium phosphate (TSP) in the containment baskets.

Date of individual notice in Federal Register: September 11, 1996 (61 FR 47962), as corrected September 26, 1996 (61 FR 50535).

Expiration date of individual notice: October 11, 1996

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

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

Date of application for amendment: August 23, 1996

Brief description of amendment request: The proposed amendment would revise Paragraph 2.B(2) of *Facility Operating License No.* DPR-40 to allow source materials in the form of depleted or natural uranium as reactor fuel and to revise Technical Specification 4.3.2 to include depleted uranium in describing the reactor core.

Date of individual notice in Federal Register: August 30, 1996 (61 FR 45995)

Expiration date of individual notice: September 30, 1996

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

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

Date of application for amendment: September 19, 1996

Brief description of amendment request: The proposed amendments would change Technical Specification requirements related to the low temperature overpressure protection (LTOP) system. Specifically, the reactor coolant system (RCS) temperature below which LTOP is required to be enabled and one high pressure safety injection pump is required to be rendered inoperable would be changed from 275 °F to 355 °F. Also, a specification would be added stating that only one reactor coolant pump shall be operated when the RCS temperature is less than or equal to 125 °F. Finally, editorial changes would be made to rename the ''Overpressure Mitigating System'' as the ''Low Temperature Overpressure Protection System.'' Date of individual notice in Federal Register: October 1, 1996 (61 FR 51308) Expiration date of individual notice: October 31, 1996

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

Wisconsin Public Service Corporation, Docket No. 50-305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: September 27, 1996

Brief description of amendment request: The proposed amendment would change Technical Specification (TS) requirements related to the low temperature overpressure protection (LTOP) system. Specifically, the LTOP curve would be modified to define 10 CFR Part 50, Appendix G pressure temperature limitations for LTOP evaluation through the end of operating cycle (EOC) 33. In addition, the LTOP enabling temperature and the temperature required for starting a reactor coolant pump would be changed consistent with the design basis for the LTOP system. Finally, the TS bases would be changed consistent with he changes described above.

Date of individual notice in Federal Register: October 7, 1996 (61 FR 52472)

Expiration date of individual notice: November 6, 1996

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, Wisconsin 54311-7001

Notice Of Issuance Of Amendments ToFacility Operating Licenses

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

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

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

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

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

Date of application for amendment: July 19, 1996

Brief description of amendment: The amendment revises the containment spray nozzle surveillance interval in TS 3/4.6.2 from 5 to 10 years.

Date of issuance: October 3, 1996

Effective date: October 3, 1996

Amendment No.: 67

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

Date of initial notice in Federal Register: August 28, 1996 (61 FR 44354) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 3, 1996.No significant hazards consideration comments received: No

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

Entergy Operations, Inc., Docket Nos. 50-313 and 50-368, Arkansas Nuclear One, Unit Nos. 1 and 2, Pope County, Arkansas

Date of amendment request: April 11, 1996, as supplemented August 23, 1996

Brief description of amendments: The amendments revised the Technical Specifications to permit implementation of 10 CFR Part 50, Appendix J, Option B.

Date of issuance: October 3, 1996

Effective date: October 3, 1996

Amendment Nos.: 185 and 176

Facility Operating License Nos. DPR-51 and NPF-6: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20846) The additional information contained in the supplemental letter dated August 23, 1996, was clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 3, 1996.No significant hazards consideration comments received: No.

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

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

Date of amendment request: April 29, 1996

Brief description of amendment: The amendment relocated cycle specific operating parameters from the Technical Specifications to the Core Operating Limits Report per Generic Letter 88-16. The parameters being relocated by this amendment include the variable low reactor coolant system pressure trip and the variable low reactor coolant system pressure-temperature protective limits.

Date of issuance: October 3, 1996

Effective date: October 3, 1996

Amendment No.: 186

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

Date of initial notice in Federal Register: June 5, 1996 (61 FR 28613) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 3, 1996.No significant hazards consideration comments received: No.

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

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

Date of amendment request: November 7, 1995, as supplemented by letter dated April 11, 1996.

Brief description of amendment: The amendment modifies the Appendix A Technical Specifications related to Safety Injection Tank level and pressure setpoints.

Date of issuance: September 27, 1996

Effective date: September 27, 1996

Amendment No.: 121

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

Date of initial notice in Federal Register: November 27, 1995 (60 FR 58401) The additional information contained in the supplemental letter dated April 11, 1996, was clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 27, 1996.No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 17, 1996

Brief description of amendments: The amendments consist of changes to the Technical Specifications regarding containment leakage tests.

Date of issuance: October 4, 1996

Effective date: October 4, 1996

Amendment Nos.: 192 and 186Facility Operating Licenses Nos. DPR-31 and DPR-41: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 28, 1996 (61 FR 44357)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 4, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendments: May 21, 1996

Brief description of amendments: The amendments revise the condensate storage tank level indication to ensure that the water level is sufficient to provide 50,000 gallons of water for core spray makeup to the reactor pressure vessel. On September 24, 1996, based on a teleconference between the licensee and the NRC project manager, it was mutually agreed to change the requested implementation schedule from 90 days to 30 days.

Date of issuance: October 2, 1996

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

Amendment Nos.: 202 and 143

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

Date of initial notice in Federal Register: August 28, 1996 (61 FR 44358) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 2, 1996. No significant hazards consideration comments received: No

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

GPU Nuclear Corporation, Docket No. 50-320, Three Mile Island Nuclear Station, Unit No. 2, (TMI-2), Dauphin County, Pennsylvania

Date of application for amendment: January 16, 1995

Brief description of amendment: This amendment revised the Technical Specification to incorporate an improvement from administrative controls section of the revised standard TS for B&W plants.

Date of issuance: October 8, 1996

Effective date: October 8, 1996

Amendment No.: 50Possession-Only License No. DPR-73: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: December 20, 1995 (60 FR 65679). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 8, 1996No significant hazards consideration comments received: No.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105

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

Date of application for amendment: July 5, 1996

Brief description of amendment: The amendment will support the implementation of noble metal chemical addition at the Duane Arnold Energy Center as a method to enhance the effectiveness of hydrogen water chemistry in mitigating intergranular stress corrosion cracking in reactor vessel internal components. Specifically, the amendment will permit an increase of the reactor water conductivity limit in Technical Specification (TS) Table 3.6.B.2-1 and several other changes in TS sections 4.6.B.2.c, 4.6.B.2.d, and the associated Bases.

Date of issuance: October 3, 1996

Effective date: October 3, 1996

Amendment No.: 218

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

Date of initial notice in Federal Register: July 31, 1996 (61 FR 40020) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 3, 1996. 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

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

Date of application for amendment: December 22, 1995, as supplemented September 20, 1996

Brief description of amendment: The amendment revises the Duane Arnold Energy Center (DAEC) Technical Specifications (TS) Sections 3.7.A and 4.7.A, ''Primary Containment,'' by deleting information also contained in 10 CFR Part 50, Appendix J, Option A and incorporating references to the Primary Containment Leakage Rate Testing Program. These changes allow the use of the performance based option of containment leak testing. The amendment also adds Operability and Surveillance Requirements (SRs) for the drywell air lock. Minor administrative changes were also made. These changes are consistent with comparable specifications in the Improved Standard Technical Specifications (ITS), NUREG-1433. In addition, the staff executed administrative changes and corrections to the TS Bases, as submitted in two letters dated February 13, 1995. Sections changed or corrected are Section 1.2, Bases; Section 2.2, Bases Reactor Coolant System Integrity; Section 3.7.H/4.7.H, Bases Containment Atmosphere Dilution; and Section 3.7.I/4.7.I, Bases Oxygen Concentration.

Date of issuance: October 4, 1996

Effective date: October 4, 1996

Amendment No.: 219

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

Date of initial notice in Federal Register: January 31, 1996 (61 FR 3499) The September 20, 1996, submittal was clarifying in nature and did not affect the no significant hazards determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 4, 1996.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

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

Date of application for amendment: June 28, 1996 and as supplemented on September 17, 1996

Brief description of amendment: The amendment will allow removal of the Inclined Fuel Transfer System (IFTS) primary containment blind flange while primary containment is required to be operable. This will provide flexibility to operate the IFTS for the purpose of testing and exercising the system during such conditions. Primary containment integrity will be provided by an alternate means while the blind flange is removed. The change will be incorporated via a provisional note into Technical Specification (TS) Surveillance Requirement 3.6.1.3.3, associated with TS 3.6.1.3, ''Primary Containment Isolation Valves (PCIVs).''

Date of issuance: October 3, 1996

Effective date: October 3, 1996

Amendment No.: 107

Facility Operating License No. NPF-62: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 31, 1996 (61 FR 40021) The information provided in the licensee's letter of September 17, 1996 provided clarifying information and did not involve significant changes to the original Federal Register notice.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 3, 1996.No significant hazards consideration comments received: No

Local Public Document Room location: The Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727

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

Date of application for amendment: June 21, 1996, and as supplemented by letter dated August 15, 1996

Brief description of amendment: The amendment modifies Section 5.7, ''High Radiation Areas,'' of the ''Administrative Controls'' section of the Clinton Power Station technical specifications (TS). The changes include: (1) allowing utilization of a Radiation Work Permit (RWP) ''or equivalent'' to control entry into a high radiation area; (2) clarifying the example given in the TS of individuals who are qualified in radiation protection procedures; (3) clarifying the requirements for when specified access controls and barriers for high radiation areas within large areas like the containment may be established; (4) clarifying that it is acceptable for an RWP to specify a maximum dose, i.e., a specified setpoint on an alarming dosimeter in lieu of a stay time for entry into a high radiation area (where an individual could receive a deep dose equivalent of 3000 mrem in one hour); (5) eliminating the upper dose limit for specifying the applicability of the requirements of Specification 5.7.1; (6) providing additional flexibility regarding the control of keys to locked doors for preventing unauthorized entry into high radiation areas; (7) providing alternate means of informing individuals of dose rates in immediate work areas; (8) reorganizing TS Sections 5.7.1, 5.7.2, and 5.7.3 into four sections (5.7.1, 5.7.2, 5.7.3 and 5.7.4); and (9) making minor edits to enhance readability.

Date of issuance: October 3, 1996

Effective date: October 3, 1996

Amendment No.: 108

Facility Operating License No. NPF-62: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 31, 1996 (61 FR 40021) The August 21, 1996, submittal consisted of supporting technical information which did not change the staff's initial proposed no significant hazards consideration determination or expand the scope of the original notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 3, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: The Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727

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

Date of application for amendment: May 2, 1996, as supplemented by letter dated August 30, 1996

Brief description of amendment: The amendment removes Technical Specification Figure 5.1, which was used in maintaining K_{eff} values, and substitutes in its place a defined requirement for maximum $K_{infinity}$ for any fuel placed in the Millstone Unit 1 spent fuel pool.

Date of Issuance: October 4, 1996

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

Amendment No.: 97

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

Date of initial notice in Federal Register: July 17, 1996 (61 FR 37301) The August 30, 1996, letter provided additional, clarifying information that did not change the scope of the May 2, 1996, application and the initial proposed no significant hazards consideration determination.The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated October 4, 1996.No significant hazards consideration comments received: No.

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

Northern States Power Company, Docket No. 50-282, Prairie Island Nuclear Generating Plant, Unit No. 1, Goodhue County, Minnesota

Date of application for amendment: July 15, 1996, and supplemented August 22, 1996

Brief description of amendment: The amendment allows the use of the moveable in-core detector system for measurement of the core peaking factors with less than 75 percent and greater than or equal to 50 percent of the detector thimbles available. The amendment is a one-time only change for Prairie Island, Unit 1, to reduce the number of required in-core detectors necessary for continued operation for the remainder of Operating Cycle 18.

Date of issuance: October 10, 1996

Effective date: October 10, 1996, and shall remain effective for the remainder of Cycle 18 only

Amendment No.: 124

Facility Operating License No. DPR-42. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 31, 1996 (61 FR 40024) By letter dated August 22, 1996, NSP forwarded a copy of the results of its most recent low power physics tests to the NRC for use as a reference and provided additional clarifying information. This information was within the scope of the original application and did not change the staff's initial proposed no significant hazards considerations determination. Therefore, renoticing was not warranted.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 10, 1996. No significant hazards consideration comments received: No.

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

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

Date of amendment request: May 17, 1996

Brief description of amendment: The amendment revises Technical Specifications (TS) 2.18, 3.14, 3.3, and 5.10 to relocate the operability requirements for shock suppressors (snubbers) from the TS to the Updated Safety Analysis Report (USAR) and incorporate snubber examination and testing requirements in TS 3.3.

Date of issuance: September 27, 1996

Effective date: September 27, 1996

Amendment No.: 176

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

Date of initial notice in Federal Register: August 28, 1996 (61 FR 44360) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 27, 1996.No significant hazards consideration comments received: No.

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

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

Date of amendment request: August 23, 1996

Brief description of amendment: The amendment modifies paragraph 2.B.(2) of

Facility Operating License No. DPR-40 allowing the use of source material, in the form of depleted or natural uranium, as reactor fuel.

Date of issuance: October 2, 1996

Effective date: October 2, 1996

Amendment No.: 177

Facility Operating License No. DPR-40: Amendment revised the Operating License.

Date of initial notice in Federal Register: August 30, 1996 (61 FR 45995) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 1996.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: January 25, 1996

Brief description of amendment: The amendment would extend the instrumentation surveillance test intervals to support 24-month operating cycles. These proposed changes would eliminate the mid-cycle outages to perform the Technical Specification surveillance requirements.

Date of issuance: October 2, 1996

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

Amendment No.: 233

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

Date of initial notice in Federal Register: May 22, 1996 (61 FR 25709) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendment: March 27, 1996, as supplemented April 24, 1996, and August 15, 1996

Brief description of amendment: The proposed amendment changes would permit implementation of 10 CFR Part 50, Appendix J, Option B with an exception to the guidelines of Regulatory Guide 1.163 for Type C testing of primary containment isolation valves in the reverse (non-accident) direction.

Date of issuance: October 4, 1996

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

Amendment No.: 234

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

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20855) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 4, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendment: August 9, 1996, as supplemented September 17, 1996

Brief description of amendment: The amendment revises the Technical Specifications to revise the safety limit minimum critical power ratio for cycle 19 operation from its current value of 1.07 (for the fuel currently in the reactor for cycle 18) for two recirculation loop operation to 1.10, and from 1.08 to 1.12 for single recirculation loop operation.

Date of issuance: October 4, 1996

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

Amendment No.: 150

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

Date of initial notice in Federal Register: August 28, 1996 (61 FR 44364) The September 17, 1996, letter provided clarifying information that did not change the scope of the August 9, 1996, application and initial proposed no significant hazards consideration determination.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 4, 1996.No significant hazards consideration comments received: No

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

South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50-395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of application for amendment: April 16, 1996, as supplemented July 25, 1996

Brief description of amendment: The amendment permits implementation of 10 CFR Part 50, Appendix J, Option B, ''Performance-Based Requirements.''

Date of issuance: October 2, 1996

Effective date: October 2, 1996

Amendment No.: 135

Facility Operating License No. NPF-12: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: July 3, 1996 (61 FR 34898) The July 25, 1996, supplement provides clarifying information and did not change the scope of the initial notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 1996.No significant hazards consideration comments received: No

Local Public Document Room location: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180

Southern California Edison Company, et al, Docket No. 50-206, San Onofre Nuclear Generating Station, Unit No. 1, San Diego County, California

Date of application for amendment: March 13, 1996

Brief description of amendment: The change revises the San Onofre Unit 1 License Condition 2.D. This change eliminates a reporting requirement that is redundant to reporting requirements in 10 CFR 50.72 and 50.73. Additionally, the amendment makes administrative and editorial changes to the Permanently Defueled Technical Specifications.

Date of issuance: October 3, 1996

Effective date: October 3, 1996

Amendment No.: 158

Facility Operating License No. DPR-13: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 31, 1996 (61 FR 40028) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 3, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Science Library, University of California, Irvine, California 92713

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 6, 1995, as supplemented by letters dated August 30, 1996, and September 20, 1996

Brief description of amendments: These amendments revise Technical Specifications (TS) Section 4.3 ''Fuel Storage'' to allow fuel assemblies having a maximum U-235 enrichment of 4.8 weight percent (w/o) to be stored in both the spent fuel racks and the new fuel racks. Additionally, TS Section 3.7.18 ''Spent Fuel Assembly Storage,'' Figures 3.7.18-1 ''Unit 1 Fuel Minimum Burnup vs. Initial Enrichment for Region II Racks,'' and 3.7.18-2 ''Units 2 and 3 Fuel Minimum Burnup vs. Initial Enrichment for Region II Racks,'' are being revised and relabeled.

Date of issuance: October 3, 1996

Effective date: October 3, 1996, to be implemented within 30 days as of the date of issuance.

Amendment Nos.: Unit 2 - 131; Unit 3 - 120

Facility Operating License Nos. NPF-10 and NPF-15: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: April 10, 1996 (61 FR 15997) The August 30, 1996, and September 20, 1996, letters provided additional clarifying information and did not change the initial no significant hazards consideration determination.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 3, 1996.No significant hazards consideration comments received: No.

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

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

Date of amendment request: July 31, 1996 (TXX-96433)

Brief description of amendments: The amendments revised core safety limit curves (Technical Specification (TS) Figure 2.1-1a) and new N-16 setpoint values and parameters (TS Table 2.1-1) for Unit 1, and reference to topical report RXE-95-001-P as an approved methodology for small break loss of coolant accident analysis for Units 1 and 2.

Date of issuance: September 30, 1996

Effective date: September 30, 1996, to be implemented within 30 days

Amendment Nos.: 52 and 38

Facility Operating License Nos. NPF-87 and NPF-89. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 28, 1996 (61 FR 44362) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 30, 1996.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 12, 1996, as supplemented by letters dated August 2, 1996, August 19, 1996, and September 5, 1996.

Brief description of amendment: The amendment revises the Technical Specifications to address the installation of laser welded tube sleeves in the Callaway Plant steam generators.

Date of issuance: October 1, 1996

Effective date: October 1, 1996, and will be implemented within 30 days of the date of issuance.

Amendment No.: 116

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

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20857) The August 2, 1996, August 19, 1996, and September 5, 1996, supplemental letters provided clarifying information and did not change the original no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluationdated October 1, 1996.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 17, 1996, as supplemented by letters dated July 15, 1996, July 31, 1996, and August 28, 1996.

Brief description of amendment: The amendment would change Technical Specification (TS) 3/4.3 to support a future modification to replace existing digital portions of the main steam and feedwater isolation system (MSFIS) with digital processor equipment and would authorize revision of the FSAR to include a description of the MSFIS modification.

Date of issuance: October 1, 1996

Effective date: October 1, 1996, to be implemented prior to startup from the Callaway Plant Refuel 8.

Amendment No.: 117

Facility Operating License No. NPF-30: The amendment revised the Technical Specifications and the Final Safety Analysis Report.

Date of initial notice in Federal Register: June 5, 1996 (61 FR 28619) The July 15, 1996, July 31, 1996 and August 28, 1996 supplemental letters provided additional clarifying information and did not change the staff's original no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 1, 1996.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: April 4, 1996

Brief description of amendment: The amendment revises the Technical Specifications regarding the surveillance requirement for control rod over-travel by moving the specific testing methodology to licensee administratively controlled documents. Specifically, the amendment removes the requirement in Specification 4.3.B.1(b) to verify prior to coupling that the over-travel indicating light is working properly by withdrawing an uncoupled control rod drive to the over-travel position.

Date of issuance: September 30, 1996

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

Amendment No.: 149

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

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20860) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 30, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendment: August 9, 1996

Brief description of amendment: The amendment changes the operations manager qualification requirements to allow either of two alternatives (having held a senior reactor operator's license or having been certified for equivalent senior reactor operator knowledge) to the requirement for the operations manager to hold a senior reactor operator's license.

Date of issuance: October 1, 1996

Effective date: October 1, 1996

Amendment No.: 148

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

Date of initial notice in Federal Register: August 28, 1996 (61 FR 44350) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 1, 1996.No significant hazards consideration comments received: No.

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

Wisconsin Public Service Corporation, Docket No. 50-305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: July 3, 1996, as supplemented on July 23, August 28, and September 16, 1996

Brief description of amendment: The amendment revises Kewaunee Nuclear Power Plant Technical Specification 4.2.b, ''Steam Generator Tubes,'' and its associated basis, by revising the acceptance criteria for indications of tube degradation occurring in the tubesheet crevice region.

Date of issuance: October 2, 1996

Effective date: October 2, 1996

Amendment No.: 129

Facility Operating License No. DPR-43: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 31, 1996 (61 FR 40031) The July 23, August 28, and September 16, 1996, submittals 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 2, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, Wisconsin 54311-7001

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: May 29, 1996, as supplemented August 20, 1996

Brief description of amendments: These amendments revise Technical Specification (TS) Section 15.4.4, ''Containment Tests,'' to incorporate the provisions of 10 CFR Part 50, Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,'' Option B. Revisions have also been made to TS Sections 15.1, ''Definitions,'' 15.3.6, ''Containment System,'' and 15.6, ''Administrative Controls,'' to support the proposed changes to Section 15.4.4.

Date of issuance: October 9, 1996

Effective date: October 9, 1996, to be implemented within 45 days.

Amendment Nos.: Unit 1 - 169 and Unit 2 - 173

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

Date of initial notice in Federal Register: July 3, 1996 (61 FR 34901) The supplemental information did not affect the staff's initial no significant hazards consideration determination.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 9, 1996.No significant hazards consideration comments received: No

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

Notice Of Issuance Of Amendments To Facility Operating Licenses And Final Determination Of No Significant Hazards Consideration And Opportunity For A Hearing (Exigent Public Announcement Or Emergency Circumstances)

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

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

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

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

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

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By November 22, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (Project Director): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-001, 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).

Duke Power Company, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of application for amendments: September 21, 1996

Brief description of amendments: The amendments approve changes to the Updated Final Analysis Report (UFSAR), and require that the changes be submitted with the next update of the UFSAR pursuant to 10 CFR 50.71(e). The associated Safety Evaluation delineates the staff's review and findings, including finding that the as-built condition of the subject power system protective devices is acceptable as-is.

Date of issuance: September 28, 1996

Effective date: September 28, 1996

Amendment Nos.: 153 and 145

Facility Operating License Nos. NPF-35 and NPF-52: The amendments revised the Updated Final Safety Analysis Report. Public comments requested as to proposed no significant hazards consideration: Yes. The NRC staff published a public notice of the proposed amendments, issued a proposed finding of no significant hazards consideration, and requested that any comments on the proposed no significant hazards consideration be provided to the staff no later than 5:00 p.m., September 28, 1996. The notice was published in ''The Herald'' of Rock Hill, South Carolina, from September 25 through 27, 1996. No comments have been received.

The Commission's related evaluation of the amendments, finding of exigent circumstances, consultation with the State of South Carolina, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated September 28, 1996.

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

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

NRC Project Director: Herbert N. Berkow

PECO Energy 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: March 25, 1996 as supplemented by letters dated August 23, 1996 and September 27, 1996.

Brief description of amendment: The amendment revises Peach Bottom Technical Specification 2.1.1.2 safety limit minimum critical power ratios to be consistent with the use of GE-13 fuel in the Unit 2 core for operating cycle 12.

Date of issuance: September 27, 1996

Effective date: As of date of issuance

Amendment No.: 217

Facility Operating License No. DPR-44: Amendment revised the Technical Specifications.Public comments requested as to proposed no significant hazards consideration: Yes (61 FR 45997). That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided an opportunity to request a hearing by September 30, 1996, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment.The Commission's related evaluation of the amendment, finding of exigent circumstances, and final no significant hazards consideration determination are contained in a Safety Evaluation dated September 27, 1996.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. Vice President and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

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

Dated at Rockville, Maryland, this 16th day of October 1996.

For the Nuclear Regulatory Commission

John A. Zwolinski,

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

[FR Doc. 96-27025 Filed 10-22-96; 8:45 am]

BILLING CODE 7590-O1-F

[Docket Nos. 50–440 and 50–346]

Perry Nuclear Power Plant, Unit 1 Davis-Besse Nuclear Power Station, Unit 1 Issuance of Director's Decision Under 10 CFR § 2.206

Notice is hereby given that the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission (NRC), has issued the Director's Decision concerning the petition dated January 23, 1996, filed by David R. Straus, Esq., et al., on behalf of the City of Cleveland, Ohio, which owns and operates Cleveland Public Power (CPP or the City) for allegedly violating the antitrust license conditions applicable to the Perry Nuclear Power Plant, Unit 1, and the Davis-Besse Nuclear Power Station, Unit 1. Supplements to the Petition were filed on May 31 and August 13, 1996.

After consideration and careful review of the facts available to the staff and the decisions reached in parallel proceedings involving the same parties and similar issues before the Federal Energy Regulatory Commission (FERC), the Director has determined that the issues raised by the petitioner that could be remedied by the NRC have been addressed and resolved in the FERC proceedings so as to require no further action by the NRC. As a result, no proceeding in response to the Petition will be instituted. The reasons for this decision are explained in the ''Director's Decision under 10 CFR § 2.206,'' (DD–96–15).

A copy of the Director's Decision has been filed with the Secretary of the Commission for Commission review in accordance with 10 CFR § 2.206(c). The Decision will become the final action of the Commission 25 days after issuance, unless the Commission on its own motion institutes review of the Decision within that time as provided in 10 CFR § 2.206(c).

Copies of the Petition, dated January 23, 1996, as supplemented May 31 and August 13, 1996, and the Notice of Receipt of Petition for Director's Decision under 10 CFR § 2.206 that was published in the Federal Register on March 8, 1996 (61 FR 9506), and other documents related to this Petition are available in the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for Perry Nuclear Power Plant (Perry Public Library, 3753 Main Street, Perry, Ohio) and Davis-Besse Nuclear Power Station (Government Documents Collection, William Carlson Library (Depository), University of Toledo, 2801 West Bancroft Avenue, Toledo, Ohio).

For the Nuclear Regulatory Commission.