For the Nuclear Regulatory Commission.

John W. N. Hickey,

Chief, Low-Level Waste and Decommissioning Projects Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards.

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NUCLEAR REGULATORY COMMISSION

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from March 29, 1997, through April 11, 1997. The last biweekly notice was published on April 9, 1997 (62 FR 17223).

Notice of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunith 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 May 23, 1997, 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: March 17, 1997

Description of amendment request: The proposed change would revise eight specifications for 18-month tests to delete a conditional statement that the testing be done while the unit is shut down and to clarify that Harris Nuclear Plant (HNP) may take credit for tests on some components which are performed while the unit is at power.

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

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

The proposed changes permit HNP to evaluate the conditions required to safely perform a test, but the changes do not directly affect the functioning or operation of any plant equipment. Since no equipment operation is involved there is no increase in the probability or consequence of any previously identified accident.

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

The proposed changes to the conditional statements on the surveillance frequencies do not involve any physical alterations or additions to plant equipment or alter the manner in which any safety-related system performs itsfunction or is operated. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes to the conditional statements on the surveillance frequency allows HNP to evaluate the conditions needed to safely perform the required testing. There is no change in the frequency of testing or in the testing which is required. There is no change in the responsibility of HNP toperform tests in a safe and responsible manner, and any changes to procedures will have to be individually evaluated to ensure that they do not reduce the margin of safety.

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

Local Public Document Room location: 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: Mark Reinhart, Acting

Commonwealth Edison Company, Docket Nos. STN 50-454 and STN 50-455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois Docket Nos. STN 50-456 and STN 50-457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of amendment request: January 30, 1997

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 1.0, ''Definitions;'' TS 3/4.6.1, ''Primary Containment'' and associated Bases; and TS 5.4.2, ''Reactor Coolant System Volume'' for Byron and Braidwood to support steam generator replacement. ComEd will be replacing the original Westinghouse D4 steam generators at Byron and Braidwood with Babcock and Wilcox International steam generators. The replacement steam generators increase the Reactor Coolant System volume which results in a higher calculated peak containment pressure (P_a) value.

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

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

Each of the [replacement steam generators] RSGs has a larger [reactor coolant system] RCS side volume than the original steam generators (OSGs). As a result of the RCS volume increase, the mass and energy release during the blowdown phase of the large break loss of coolant accident (LBLOCA) is increased. Additionally, the heat transfer rate of the RSGs is greater than the OSGs, and the RSGs will operate at a slightly higher pressure than that for the OSGs. Consequently, the steam enthalpy exiting the break during the reflood period, with the RSG, will be greater than that for the OSG. This results in an increase in the containment building peak pressure, P_a.

The proposed revisions to the Technical Specifications involve the specified value of Unit 1 RCS volume and the defined value of Unit 1 P_a. Several editorial changes are also being made to improve clarity and consistency of the TS.

RCS volume is not an initiator for any event and an increase in volume does not affect any operating margin or requirements. Therefore, increasing the primary volume does not increase the probability of any event previously analyzed.

The revised value of P_a continues to be less than the design basis pressure for the containment building structure. The change represents only a revision to the containment test pressure for containment leakage testing. Such testing is only performed with the affected unit in the shutdown condition. Therefore, the proposed change in P_a does not involve a significant increase in the probability of an accident previously evaluated.

All accidents in the Updated Final Safety Analysis Report (UFSAR) were evaluated to determine the effect of an increase in primary volume on accident consequences. The events identified that may be impacted by an increase in primary volume are the Waste Gas System Leak or Failure and LBLOCA. For the Waste Gas System Leak or Failure, the activity of the decay tank is controlled to Technical Specification limits which are unaffected by RCS volume. Therefore, an increase in RCS volume would not increase the offsite dose.

The offsite dose calculation for the LBLOCA is unaffected by the proposed change. The license basis offsite dose calculation is in accordance with NRC Reg Guide 1.4 ''Assumptions Used for Evaluating The Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors.'' This Regulatory Guide states, in part, ''...a number of appropriately conservation assumptions, based on engineering judgment and on applicable experimental results from safety research programs conducted by the AEC.'' These conservatisms include (but are not limited to) the following assumptions:

- Twenty five percent of the equilibrium radioactive full power inventory is immediately available for leakage from the primary containment.
- 100% of the equilibrium full power radioactive noble gas inventory is immediately available for leakage from the primary containment.
- The primary containment should be assumed to leak at the (maximum) leak rate specified in the technical specifications for the first 24 hours and at 50% of this value for the remaining 29 days of the accident duration.

The design basis leakage corresponding to a peak containment pressure of 50 psig utilized in the design basis accident analysis is 0.10% per day of the containment free air mass. Therefore, the offsite dose calculation was performed with a leakage of .1% per day for day one and .05% per day for days two through 30. Isotopic inventories are unaffected by the increase in reactor coolant volume. Thus, the offsite dose is unaffected by the increase in the peak containment pressure. Therefore, this proposed change to P_a does not involve a significant increase in the consequences of an accident previously evaluated.

The editorial changes proposed are for clarity and consistency within the Technical Specifications and do not affect either the probability or consequences of an accident previously evaluated.

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

The proposed change in RCS volume is a change in a plant parameter within the ''Design Features'' section of the Technical Specifications. Increasing the RCS volume does not create any new or different failure modes. The existing RCS design requirements continue to be met.

The revised value of P_a continues to be less than the design basis pressure for the containment building structure. The change represents only a revision to the test pressure for containment leakage testing. Such testing is only performed with the affected unit in the shutdown condition. Therefore, no new or different failure modes are being introduced by modification of the testing parameters.

The editorial changes proposed are for clarity and consistency within the Technical Specifications and do not result in any physical changes to the facility or how it is operated. No new or different failure modes are being introduced by these changes.

Therefore, these proposed changes do 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.

Changing the RCS volume in the Technical Specifications does not reduce the margin of safety. RCS volume is a design feature. The change in RCS volume does not involve a change to any setpoint or design requirements. An evaluation of all UFSAR accidents was performed to determine the effect of an increase in RCS volume. This evaluation is summarized as follows:

An evaluation of the Chemical and Volume Control System Malfunction was performed to determine the effect of the increased RCS volume due to the RSGs. The larger RCS volume of the RSGs reduces the reactivity insertion for a given dilution flow rate. Therefore, the UFSAR analyses remain bounding for Byron Unit 1 and Braidwood Unit 1 with the RSGs and there is no reduction in the margin of safety.

An evaluation of the Inadvertent Actuation of the Emergency Core Cooling System During Power Operation Event was performed to determine the effect of the increased RCS volume due to the RSGs. For this event, the injection of borated water causes a negative reactivity insertion, which increases DNBR. For a given Refueling Water Storage Tank (RWST) boron concentration, the larger RCS volume will cause a reduction in the negativity insertion rate as compared to the current UFSAR analysis. However, negative reactivity would still be inserted, no fuel pins would experience DNB, and there is no reduction in the margin of safety.

An evaluation of the Small Break LOCA was performed to determine the effect of increased RCS volume. The additional RCS volume will cause a delay in the loop seal clearing which in turn delays the core uncovery as compared with the UFSAR analysis. A delay in core uncovery reduces the amount of core heatup which results in a lower peak clad temperature (PCT) because the core decay heat would be less than in the UFSAR analysis. The benefit is considered small, but there is still a benefit. Therefore, the increased RCS volume does not result in a reduction in the margin of safety.

An evaluation of the Large Break LOCA was performed to determine the effect of increased RCS volume. For a LBLOCA, the increased RCS volume causes the blowdown phase of the event to be longer. Increased blowdown phase, alone, could potentially result in a higher PCT. However, the RSGs also have less resistance to flow due to increased primary side steam generator flow area, which results in a higher blowdown flow compared to the OSGs. The increased blowdown flow more than compensates for the longer blowdown phase associated with the increased RCS volume. The net effect is a decrease in PCT for the RSG compared to the OSG. Therefore, there is no reduction in the margin of safety.

An evaluation of the Gas Waste System Leak or Failure was performed to determine the effect of the increased RCS volume. Because the activity of the decay tank is controlled within Technical Specification limits, an increase in RCS volume would not change the results of the event. Therefore, there is no reduction in the margin of safety.

An evaluation was performed to determine the effect of the increased RCS volume on the peak containment pressure following a LBLOCA. The increased RCS volume caused the peak containment pressure to increase to 47.8 psig. This is still below the containment design pressure of 50.0 psig. Therefore, there is no reduction in the margin of safety.

This proposed change involves testing requirements designed to demonstrate adequate leakage rates are maintained. If adequate leakage rates are maintained as outlined in the Technical Specifications, there will be no reduction in the margin of safety. In the event of degradation of a containment seal that results in unacceptable leakage, plant shutdown will occur as required by Technical Specifications and administrative requirements in accordance with approved plant procedures. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

The editorial changes proposed are for clarity and consistency within the Technical Specifications and do not result in any physical changes to the facility or how it is operated. Therefore, the changes have no effect on the margin of safety.

Thus, this amendment request does not result in any decrease 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 requested amendments involve no significant hazards consideration.

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

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

NRC Project Director: Robert A. Capra

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

Date of amendment request: March 27, 1997

Description of amendment request: The proposed amendment would alter the company name in the Facility Operating License DPR-20 and Technical Specifications for the Palisades Plant. Specifically, the proposed amendment would revise the name from ''Consumers Power Company'' to ''Consumers Energy Company.''

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

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

Since the proposed changes do not alter the technical content of any Facility Operating License or Technical Specifications requirements, they do not alter any feature of plant equipment, settings, operation, or configuration.

Therefore, they cannot involve a significant increase in the probability of an accident previously evaluated.

The proposed changes alter the company name in the Facility Operating License and Technical Specifications to reflect the change from ''Consumers Power Company'' to ''Consumers Energy Company''. The proposed change will not affect any obligations. The company will continue to own all of the same assets, will continue to serve the same customers, and will continue to honor all existing obligations and commitments. The proposed changes will not alter plant operation or configuration, or its ability to respond to accidents.

Therefore, they will not involve a significant increase in the consequences of any accident previously evaluated.

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

Since the proposed changes do not alter the technical content of any Facility Operating License or Technical Specifications requirements, they do not alter any feature of plant equipment, settings, operation or configuration.

Therefore, they cannot create the possibility of a new or different kind of accident from any previously evaluated.

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

Since the proposed changes do not alter the technical content of any Facility Operating License or Technical Specifications requirements, they do not alter any feature of plant equipment, settings, operation, or configuration.

Therefore, they cannot involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John N. Hannon

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

Date of amendment request: March 31, 1997 (TSC 96-10)

Description of amendment request: The proposed amendments would modify and clarify the High Pressure Injection (HPI) System operability requirements in Specification 3.3.1, impose additional HPI system operability requirements for operation above 75 percent power, incorporate the new Standard Technical Specifications format for the HPI system, revise Specification 3.3.2 to clarify that the Reactor Building Emergency Sump isolation valves are remote-manually operated valves, and add new specifications and a surveillance test to address operability requirements of the atmospheric dump valves. In addition, corresponding Bases changes would be incorporated.

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. None of the proposed changes has any impact upon the probability of any accident which has been evaluated in the UFSAR [Updated Final Safety Analysis Report]. The only potential change in operating configuration is allowing operation with the HPI [High Pressure Injection] System pump discharge header cross-connected. This operating mode does not affect the probability of a LOCA [Loss-of-Coolant Accident] or of any other accident evaluated in the UFSAR.

None of these changes have any impact upon the ability of the HPI System to add soluble poison to the Reactor Coolant System, as addressed by Specification 3.2. The remaining potential impact is upon the ability to mitigate the consequences of a small break LOCA, which is addressed below. The small break LOCA is the limiting design basis accident with respect to HPI System operability requirements.

The proposed changes to Specification 3.3.1 provide appropriate actions to address any degradation in the operability of the HPI System. The operability requirements for the HPI System are supported by a spectrum of small break LOCA analyses based on the approved Evaluation Model described in FTI [Framatome Technologies, Incorporated] topical report BAW-10192P. These small break LOCA analyses demonstrate that the acceptance criteria of 10CFR 50.46 are not violated.

Two trains of HPI are required to mitigate a small break LOCA above 75% FP [full power]. Operability requirements in the proposed Technical Specifications assure that the HPI System can withstand the worst single failure and still result in two HPI pumps injecting through two trains. The full power small break LOCA analyses supporting this proposed license amendment have been performed in accordance with the approved Evaluation Model described in FTI topical report BAW-10192P. The proposed Technical Specifications limit operation above 75% FP with a degraded HPI System to 72 hours before a power reduction to less than 75% FP (or a reactor shutdown) must be initiated. The required actions depend on the HPI System components that are inoperable. The 72 hour completion time is consistent with the time requirements for HPI specified in NUREG-1430.

When at or below 75% FP, one HPI train provides sufficient flow to mitigate a small break LOCA. The 75% power level is justified by analyses using the Evaluation Model described in FTI topical report BAW-10192P, considering the worst case break location and size described in LER [Licensee Event Report] 269/90-15 and Attachment 3 to this submittal. The proposed Technical Specifications require two HPI trains to be operable at or below 75% FP. These requirements ensure that, following the worst single failure, one train of HPI would remain available to mitigate a small break LOCA. Operation with less than two HPI trains operable is restricted to 72 hours before shutdown requirements are imposed. This completion time is consistent with the time requirements specified for an HPI System in NUREG-1430.

The additional HPI system restriction that requires the HPI pump discharge header to be cross-connected when all three HPI pumps are operable does not increase the consequences of a small break LOCA. If a single failure prevents one HPI train from actuating, this lineup results in at least two HPI pumps initially injecting through the automatically actuating train. This increases the amount of cooling flow initially delivered to the core as compared to the current system configuration.

The impact of this alignment has been evaluated, considering the potential single active failures, including the failure of any powered component to operate and any single failure of electrical equipment.

It has been determined that, when each of the three HPI pumps is either running or is capable of automatic actuation upon an Engineered Safeguards signal, cross-connection of the HPI pump discharge header does not introduce susceptibility to any single failure. Therefore, the potential consequences of a small break LOCA are not increased. If fewer than three HPI pumps are either running or are capable of automatic actuation, and the HPI pump discharge header were cross-connected, a single failure of one pump could cause a single pump to be aligned to both HPI trains. In this condition, the single pump could experience runout conditions prior to corrective operator action. However, proposed Specification 3.3.1 requires the discharge header to be isolated between the two remaining operable HPI pumps. The proposed BASES provide guidelines to ensure that the requirements for redundancy are properly implemented. Therefore, the proposed specifications ensure that the consequences of a small break LOCA are not increased by allowing the HPI pump discharge header to be cross-connected.

In addition, proposed Specification 3.4.7 requires new operability requirements for the main steam atmospheric dump valves. These operability requirements do not impact the probability or consequences of any accident. The proposed specification for the atmospheric dump valves provides additional assurance that these valves will be operable in the event of a small break LOCA.

In summary, the proposed Technical Specifications provide adequate controls to assure that operability of the HPI System is maintained in a manner consistent with the requirements of the design basis accidents. Therefore, it is concluded that this amendment request will not significantly increase the probability or consequences of an accident previously evaluated.

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

No. Of the proposed substantive changes, only cross-connection of the HPI pump discharge header represents any change to the way in which the facility is normally operated. Operation with the discharge header cross-connected is not a new configuration, as it has always been used for HPI pump testing both at power and during shutdown conditions. Potential failure modes have already been considered as described earlier. No new initiating events or potentially unanalyzed conditions have been created. Therefore, this proposed amendment will not create the possibility of any new or different kind of accident.

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

No. The HPI restrictions associated with the proposed Technical Specifications are supported by analyses which demonstrate that the acceptance criteria of 10 CFR 50.46 are not violated for any small break LOCA. These analyses were performed in accordance with the Evaluation Model described in FTI topical report BAW-10192P. Therefore, it is concluded that the proposed amendment request will not result in a significant decrease in the margin of safety.

Duke has concluded, based on the above, that there are no significant hazards considerations involved in this amendment request.

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: 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 Nos. 50-334 and 50-412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of amendment request: March 10, 1997

Description of amendment request: The proposed amendment would modify Technical Specification 3.4.5, ''Steam Generators,'' and associated Bases to allow repair of steam generator tubes by installation of sleeves with the tungsten inert gas (TIG) welded sleeve developed by ABB Combustion Engineering. In addition, the proposed amendment would delete the option for using the kinetic sleeving methodology previously approved for use at Beaver Valley, but is not currently recommended by Framatome Technologies, Inc.

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 allows the ABB Combustion Engineering (ABB/CE) tungsten inert gas (TIG) welded tubesheet sleeves and tube support plate sleeves to be used as an alternate steam generator tube repair method. The sleeve configuration was designed and analyzed in accordance with the criteria of Regulatory Guide (RG) 1.121 and Section III of the ASME [American Society of Mechanical Engineers] Code. Fatigue and stress analyses of the sleeved tube assemblies produce acceptable results for both types of sleeves as documented in ABB/CE Topical Report CEN-629-P, Revision 02 and CEN-629-P Addendum 1. Mechanical testing has shown that the structural strength of the sleeves under normal, faulted, and upset conditions is within the acceptable limits specified in RG 1.121. Leakage rate testing for the tube sleeves has demonstrated that primary to secondary leakage is not expected during any plant condition. The consequences of leakage through the sleeved region of the tube is fully bounded by the existing steam generator tube rupture (SGTR) analysis included in the Updated Final Safety Analysis Report (UFSAR).

The sleeves are designed to allow inservice inspection of the pressure retaining portions of the sleeve and parent tube. Inservice inspection is performed on all sleeves following installation to ensure that each sleeve has been properly installed and is structurally sound. Periodic inspections are performed in subsequent refueling outages to monitor sleeve degradation on a sample basis. The eddy current technique used for inspection will be capable of detecting both axial and circumferential flaws. Specific guidance for steam generator sleeve inspection is provided in the current technical specification surveillance requirements. Tubes that contain defects in a sleeve, which exceed the repair limit, will be removed from service. This ensures that sleeve and tube structural integrity is maintained.

The proposed TS change to support the installation of TIG welded sleeves does not adversely impact any previously evaluated design basis accident. The effect of sleeve installation on the performance of the SG [steam generator] was analyzed for heat transfer, flow restriction, and steam generation capacity. The sleeves reduce the risk of primary to secondary leakage in the SG. The installation of ABB/CE sleeves results in a hydraulic flow restriction that is dependent on the number and types of sleeves installed. The reduction in primary system flow rate is a small percentage of the flow rate reduction seen from plugging one tube and is a preferable alternative when considering core margins based on minimum reactor coolant system flow rates. The sleeving installation will result in a resistance to primary coolant flow through the tube for other evaluated accidents. The results of the analyses and testing, as well as industry operating experience, demonstrate that the sleeve assembly is an acceptable means of maintaining tube integrity. In summary, installation of sleeves does not substantially affect the primary system flow rate or the heat transfer capability of the steam generators.

Installation of the sleeves can be used to repair degraded tubes by returning the condition of the tubes to their original design basis condition for tube integrity and leak tightness during all plant conditions. The tube bundle overall structural and leakage integrity will be increased with the installation of the sleeves reducing the risk of primary to secondary leakage in the SG while maintaining acceptable reactor coolant system flow rates. Therefore, sleeving will not increase the probability of occurrence of an accident previously evaluated.

Removal of the kinetically welded sleeve process as an approved SG tube repair methodology will have no effect on plant operations. There are currently no kinetically welded sleeves installed in the steam generators. Had there been, plant operations would have still been bounded by the existing SGTR analysis in the UFSAR.

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

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

The implementation of the proposed sleeving process will not introduce significant or adverse changes to the plant design basis. Stress and fatigue analyses of the repair has shown the ASME Code Section III and RG 1.121 allowable values are met. Implementation of TIG welded sleeving maintains overall tube bundle structural and leakage integrity at a level consistent with that of the originally supplied tubing. Leak and mechanical testing of the sleeves support the conclusions that the sleeve retains both structural and leakage integrity during all conditions. Repair of a tube with a sleeve does not provide a mechanism that would result in an accident outside of the area affected by the sleeve.

Any hypothetical accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing SGTR analysis. The SGTR analysis accounts for the installation of sleeves and the impact on current plugging level analyses. The sleeve design does not affect any other component or location of the tube outside of the immediate area repaired.

The current primary to secondary leakage limit ensures that SG tube integrity is maintained in the event of an MSLB [main steam line break] or LOCA [loss-of-coolant accident]. The limit will provide for leakage detection and a plant shutdown in the event of the occurrence of an unexpected single crack resulting in excessive tube leakage. The leakage limit also provides for early detection and a plant shutdown prior to a postulated crack reaching critical crack lengths for MSLB conditions.

Inservice inspections are performed following sleeve installation to ensure proper weld fusion has occurred to maintain structural integrity. The post installation inspection also serves as baseline data to be used for comparison during future inspections. Periodic eddy current inspections monitor the pressure retaining portions of the sleeve and parent tube for degradation. Eddy current techniques will be employed that are sensitive to axial and circumferential degradation.

Increasing the sample size of tubes repaired using either sleeving process during each scheduled inservice inspection will increase the monitoring of these tubes for any further degradation. The improved monitoring and evaluation of the tube and the sleeves assures tube structural integrity is maintained or the tube is removed from service.

Corrosion testing of typical sleeve-tube configurations was performed to evaluate local stresses, sleeve life, and resistance to primary and secondary side corrosion. The tests were performed on stress relieved and as-welded (non-stress relieved) sleeve-tube joints. Using the corrosion test data in conjunction with finite element analyses of the local stress, the stress relieved joint life was determined to be in excess of 40 years. The ABB/CE TIG welded sleeve operating experience in the industry has shown no sleeve failures due to service induced degradation in sleeves that were installed with acceptable inspection results. This experience includes the stress relieved and as-welded sleeve configurations. All sleeves will be stress relieved as specified in the topical report.

Removal of the kinetically welded sleeve process as an approved SG tube repair methodology and not completing the additional corrosion testing necessary to establish the design life for the kinetically welded sleeve in the presence of a crevice will not create the possibility of a new or different type of accident from any accident previously evaluated.

Repair of an SG tube with a kinetically welded sleeve would not have provided a mechanism that resulted in an accident outside of the area affected by the sleeve. Any hypothetical accident as a result of potential tube or sleeve degradation in the repaired portion of the tube would have been bounded by the existing SGTR analysis. The SGTR analysis accounts for the installation of sleeves and the impact on current plugging level analyses. The sleeve design does not affect any other component or location of the tube outside of the immediate area repaired. Furthermore, there are currently no kinetically welded sleeves installed in either plant.

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

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

The TIG welded sleeving repair of degraded steam generator tubes has been shown by analysis to restore the integrity of the tube bundle to its original design basis condition. The safety factors used in the design of the sleeves for the repair of degraded tubes are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code Section III used in steam generator design. The design of the ABB/CE SG sleeves has been verified by testing to preclude leakage during normal and postulated accident conditions.

The portion of the installed sleeve assembly which represents the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation, thus satisfying the requirement of RG 1.83. The portion of the SG tube bridged by the sleeve joints is effectively removed from the pressure boundary, and the sleeve then forms the new pressure boundary. The sleeve enhances the safety of the plant by reestablishing the protective boundaries of the steam generator. Keeping the tube in service with the use of a sleeve instead of plugging the tube and removing it from service increases the heat transfer efficiency of the steam generator. During each scheduled inservice inspection, each sleeve inspected and found to have unacceptable degradation shall be removed from service.

The effect on the design transients and the accident analyses have been revised based on the installation of sleeves equal to the tube plugging level coincident with the minimum reactor coolant flow rate. Evaluation of the installation of sleeves was based on the determination that LOCA evaluations for the licensed minimum reactor coolant flow bound the combined effect of tube plugging and sleeving up to an equivalent of the actual plugging limit. Sleeving results in a fractional amount of the plugging limitation of one tube and is a preferable alternative when considering core margins based on minimum reactor coolant system flow rates. The sleeving installation will result in a resistance to primary coolant flow through the tube. The primary coolant flow through the ruptured tube is reduced by the influence of the installed sleeve; therefore, the consequences to the public due to an SGTR event have not increased.

As SG sleeve removes an indication of a possible leak source from the reactor coolant system (RCS) pressure boundary, eliminating the potential of a primary-to-secondary leak. The structural integrity of the tube is maintained by the sleeve and sleeve-to-tube joint.

Installation of either tube sheet or tube support plate sleeves will increase the protective boundaries of the steam generators and will not reduce the margin of safety.

Removal of the kinetically welded sleeve process as an approved SG tube repair methodology will not result in a reduction in the margin of safety. There are currently no kinetically welded sleeves installed in either plant. SG tube integrity will be maintained by applying an alternate NRC approved repair methodology or removing the SG tube from service by plugging.

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

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

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

Date of amendment request: March 10, 1997

Description of amendment request: The proposed amendment would revise Technical Specifications 3.4.5, ''Steam Generators,'' and associated Bases to allow repair of steam generator tubes by installation of sleeves with the Electrosleeving process developed by Framatome Technologies, Inc. (FTI).

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 Electrosleeve configuration has been designed and analyzed in accordance with the requirements of the ASME [American Society of Mechanical Engineers] Code. The applied stresses and fatigue usage for the Electrosleeve are bounded by the limits established in the ASME Code. Minimum material property values are used for the structural and plugging limit analysis. Mechanical testing has shown that the structural strength of nickel Electrosleeves under normal, upset, and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (3 times normal operating pressure differential) burst margin recommended by Regulatory Guide 1.121. Leakage testing has shown that the Electrosleeve is essentially leaktight during all plant conditions.

The Electrosleeve nominal wall thickness depth-based plugging limit is determined using the guidance of Regulatory Guide 1.121 and the pressure stress equation of Section III of the ASME Code. The limiting requirement of Regulatory Guide 1.121 for the Electrosleeve, which applies to part through wall degradation, is the minimum acceptable wall thickness to maintain a safety factor of three against tube failure under normal operating conditions. A bounding set of design and transient loading input conditions was used for the minimum wall thickness evaluation in the generic evaluation. Evaluation of the minimum acceptable wall thickness for normal, upset and postulated accident condition loading per the ASME Code indicates these conditions are bounded by the design minimum wall thickness.

Bounding tube wall degradation growth rate per cycle and nondestructive examination uncertainty has been assumed for determining the Electrosleeve technical specification plugging limit. Electrosleeve wall degradation extent determined by nondestructive examination, which would require plugging Electrosleeved tubes, is developed using the guidance of Regulatory Guide 1.121 and is defined in FTI Topical Report BAW-10219P, Revision 1, to be 20% throughwall of the nominal sleeve wall thickness.

The effect of Electrosleeving and plugging will remain below the plugging limit assumed in the UFSAR [Updated Final Safety Analysis Report]. The proposed change will not increase the consequences of these accidents.

The results of the analyses and testing demonstrate that the Electrosleeve is an acceptable means of maintaining tube integrity. Further, per Regulatory Guide 1.83 recommendations, the Electrosleeved tube can be monitored through periodic inspections with present NDE [nondestructive examination] techniques. These measures demonstrate that installation of Electrosleeves spanning degraded areas of the tube will restore the tube to a condition consistent with its original design basis.

Since the main steamline break post-accident primary-to-secondary leakage is not increased by the presence of Electrosleeves, the consequences of an accident previously evaluated in the UFSAR are not increased. Conformance of the Electrosleeve design with the applicable sections of the ASME Code and results of the leakage and mechanical tests support the conclusion that installation of Electrosleeves does not increase 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?

Electrosleeving will not adversely affect any plant component. Stress and fatigue analysis of the repair has shown that the ASME Code and Regulatory Guide 1.121 criteria are not exceeded. Implementation of Electrosleeving maintains overall tube bundle structural and leakage integrity at a level consistent with that of the original tubing during all plant conditions. Leak and mechanical testing of Electrosleeves support the conclusions of the calculations that each Electrosleeve retains both structural and leakage integrity during all conditions. Electrosleeving of tubes does not provide a mechanism resulting in an accident outside of the area affected by the Electrosleeves. Any accident resulting from potential tube or Electrosleeve degradation in the repaired portion of the tube is bounded by the existing tube rupture accident analysis.

Implementation of Electrosleeving will reduce the potential for primary-to-secondary leakage while not significantly impacting available primary coolant flow area in the event of a LOCA. By effectively isolating degraded areas of the tube through repair, the potential for steamline break leakage is reduced. These degraded intersections now are returned to a condition consistent with the Design Basis. While the installation of an Electrosleeve reduces primary coolant flow, the reduction is far below that caused by plugging. Greater primary coolant flow area is maintained through Electrosleeving versus plugging. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

The Electrosleeve repair of degraded steam generator tubes has been shown by analysis to restore the integrity of the tube bundle consistent with its original design basis condition. The tube/Electrosleeve operational and faulted condition stresses are bounded by the ASME Code requirements and the Electrosleeved tubes are leaktight. The safety factors used in the design of Electrosleeves for the repair of degraded tubes are consistent with the safety factors in the ASME Code used in steam generator design. The portions of the installed Electrosleeve assembly which represent the reactor coolant pressure boundary can be monitored for the initiation and progression of Electrosleeve/tube wall degradation, thus satisfying the requirements of Regulatory Guide 1.83. The portion of the tube bridged by the Electrosleeve is effectively removed from the pressure boundary, and the Electrosleeve then forms the new pressure boundary. The areas of the Electrosleeved tube assembly which require inspection are defined in Framatome Technologies Inc. Topical Report BAW-10219P, Revision 1.

In addition, since the installed Electrosleeve represents a portion of the pressure boundary, a baseline inspection of these areas is required prior to operation with Electrosleeves installed. The effect of sleeving on the design transients and accident analyses has been reviewed based on the installation of Electrosleeves up to the level of steam generator tube plugging coincident with the minimum reactor coolant flow rate and UFSAR and has been found acceptable.

It is concluded that the proposed license amendment request does not result in a significant reduction in the margin of safety as defined in the UFSAR or technical specifications.

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

Local Public Document Room location: 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

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

Date of amendment request: March 27, 1997

Description of amendment request: The proposed changes to the Technical Specifications (TSs) would modify the limiting condition for operation (LCO) and surveillance requirements (SR) for the ultimate heat sink. The ultimate heat sink for Millstone Unit No. 2 is the Long Island Sound that transfers heat from safety-related systems during normal and accident conditions. Specifically, TS LCO 3.7.11 would be changed to indicate that the ultimate heat sink is operable at a water temperature of less than or equal to 75 °F instead of an average value. TS SRs 4.7.11.a and .b would also delete the use of average when verifying the water temperature and delete the reference to a specific monitoring location, the Unit No. 2 intake structure. These proposed changes do not change the ultimate heat sink temperature limit, which remains at a maximum of 75 °F.

The TS Bases 3/4.7.11 would also be modified to reflect the above changes and to identify the various locations that the ultimate heat sink temperature can be measured.

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

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

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

The proposed changes remove the reference to a monitoring location where the temperature of the ultimate heat sink is measured and eliminate the use of an average ultimate heat sink temperature. The instruments used provide information to the operators which will permit them to ensure that the plant is operated within the design basis of the plant. The subject instruments will provide an accurate representation of the ultimate heat sink temperature. This role is passive; thus, these instruments cannot initiate or mitigate any accident.

The locations used to monitor the ultimate heat sink temperature will be maintained in the Bases. This is a licensee controlled document which is maintained under the requirements of 10CFR50.59. The details being removed from the Technical Specifications are not assumed to be an initiator of any analyzed event. Since any changes to the relocated details will be evaluated per 10CFR50.59, any possible increase in the probability or consequences of an accident previously evaluated will be addressed.

The proposed changes do not revise the ultimate heat sink temperature limit of 75 °F. The current analysis is based on the ultimate heat sink temperature limit of 75 °F. Therefore, there is no effect on the consequences of any accident previously evaluated.

Thus, the license amendment request does not impact the probability of an accident previously evaluated nor does it involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes remove the reference to a monitoring location where the temperature of the ultimate heat sink is measured and eliminate the use of an average ultimate heat sink temperature. The instruments used provide information to the operators which will permit them to ensure that the plant is operated within the design basis of the plant. The subject instruments will provide an accurate representation of the ultimate heat sink temperature. This role is passive, thus, these instruments cannot initiate or mitigate any accident.

The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. They will not alter assumptions made in the safety analysis and licensing basis.

The locations used to monitor the ultimate heat sink temperature will be maintained in the Bases. This is a licensee controlled document which is maintained under the requirements of 10CFR50.59. Thus, adequate control of information will be ensured.

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

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

The proposed changes remove the reference to a monitoring location where the temperature of the ultimate heat sink is measured and eliminate the use of an average ultimate heat sink temperature. They do not change the ultimate heat sink temperature limit of 75 °F, which is assumed by the current analysis. Therefore, there is no effect on the consequences of any accident previously evaluated and no significant impact on offsite doses associated with previously evaluated accidents. Thus, there is no significant reduction in the margin of safety for the design basis accident analysis. The license amendment request does not result in a reduction of the margin of safety as defined in the Bases for Technical Specification 3.7.11. The instruments used provide information to the operators which will permit them to ensure that the plant is operated within the design basis of the plant. The subject instruments will provide an accurate representation of the ultimate heat sink temperature. The proposed changes do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. They do not have any impact on the protective boundaries (e.g., fuel matrix and cladding, reactor coolant system pressure boundary, and primary and secondary containment), or on the safety limits for these boundaries.

The locations used to monitor the ultimate heat sink temperature will be maintained in the Bases. The Bases are a licensee controlled document which is maintained under the requirements of 10CFR50.59. Since any future changes to this license controlled document will be evaluated per the requirements of 10CFR50.59, any possible reduction (significant or insignificant) in a margin of safety will be addressed.

Thus, the license amendment request does not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Learning 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 06141-0270NRC Deputy Director: Phillip F. McKee

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

Date of amendment request: March 31, 1997

Description of amendment request: The proposed amendment would modify Technical Specification Surveillance Requirement 4.7.1.2.1.b which requires the testing of the auxiliary feedwater motor-driven and turbine-driven pumps on recirculation flow at least once per 92 days. The proposed amendment would also makes changes to the appropriate Bases section.

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

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

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

The proposed changes to Technical Specification Surveillance 4.7.1.2.1.b to increase the required test parameter for the motor driven pumps from 1460 psid to 1468 psid and replacing the current parameters for the motor driven and turbine driven pumps from differential pressure measured in psid [pounds per square inch differential] to total head measured in feet are consistent with equipment design criteria and does not significantly increase the probability of an accident previously evaluated.

The proposed changes to increase the required test parameter for the motor driven pumps from 1460 psid to 1468 psid and replacing the current parameters for the motor driven and turbine driven pumps from differential pressure measured in psid to total head measured in feet provides the necessary assurance that the pumps will function as required in accident analyses and does not significantly increase the consequence of an accident previously evaluated.

The moving of the reference to Specification 4.0.5 in order to clarify that it applies to the testing of the motor driven and turbine driven pumps and the modifications to the bases section are administrative and do not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

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

The proposed changes to Technical Specification Surveillance 4.7.1.2.1.b to increase the required test parameter for the motor driven pumps from 1460 psid to 1468 psid and replacing the current parameters for the motor driven and turbine driven pumps from differential pressure measured in psid to total head measured in feet does not change the operation of the auxiliary feedwater system or any of its components during normal or accident evaluations.

The moving of the reference to Specification 4.0.5 in order to clarify that it applies to the testing of the motor driven and turbine driven pumps and the modifications to the bases section are administrative and do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

The proposed change to Technical Specification Surveillance 4.7.1.2.1.b to increase the referenced total head of the motor

driven auxiliary feedwater pumps during surveillance testing provides an acceptable margin between the required surveillance and design pump performance to provide assurance that the pumps will operate consistent with system evaluations and does not involve a significant reduction in a margin of safety.

The change in the referenced units from differential pressure measured in psid to total head measured in feet for the motor driven auxiliary and turbine driven auxiliary feedwater pumps during surveillance testing is to account for the effect of water density on pump performance during each test and does not involve a significant reduction in a margin of safety.

The moving of the reference to Specification 4.0.5 in order to clarify that it applies to the testing of the motor driven and turbine driven pumps and the modifications to the bases section are administrative and do not involve a significant reduction in a margin of safety.

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

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

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

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

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

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

Date of amendment request: March 31, 1997

Description of amendment request: The proposed amendment would separate the required testing of motor-operated valve thermal overload protection into two new surveillances.

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

NNECO has reviewed the proposed change in accordance with 10CFR50.92 and has concluded that the change does not involve a significant hazards consideration (SHC). The bases for this conclusion is that the three criteria of 10CFR50.92(c) are not satisfied. The proposed change does not involve a SHC because the change would not:

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

The proposed changes to the surveillance testing of the motor-operated valve thermal overload protection are consistent with equipment design criteria and performing surveillance testing does not significantly increase the probability of an accident previously evaluated. The proposed changes to the surveillance testing provides the necessary assurance that the motor operated valve thermal overload protection will function as required and does not involve a significant increase in the consequence of an accident previously evaluated.

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

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

The proposed changes to the surveillance testing of the motor-operated valve thermal overload protection does not change the operation of any system or system component during normal or accident evaluations.

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

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

The proposed changes to the surveillance testing of the motor-operated valve thermal overload protection are administrative in that the changes to the surveillance only clarify that following maintenance on the motor starter, a channel calibration is required only on that valve. The surveillance continues to require periodic representative sample testing.

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

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

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

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

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

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: March 4, 1997

Description of amendment request: The amendments would modify the Emergency Core Cooling System (ECCS) surveillance test acceptance criteria in Technical Specification 3/4.5.2 for the Centrifugal Charging (CH) and the Safety Injection (SI) pumps. The changes to the specified flow values would account for system alignments that effect the suction pressure to the pumps. In the recirculation mode, increased flow occurs when the CH and SI pumps take suction from the discharge of the Residual Heat Removal pumps.

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 evaluations performed by Westinghouse determined that, with the proposed changes, the subject pumps remain operable and the safety analyses criteria remain valid.

Previous conclusions under LCR [License Change Request] 91-03 evaluating the consequences of the LOCA [loss-of-coolant-accident] considered in the Salem Units 1 & 2 licensing basis remain unchanged. With respect to the LOCA, the Peak Cladding Temperature (PCT) continues to conform to the 10CFR50.46 guidelines of less than 2200*F. Evaluation of LOCA mass and energy releases previously found acceptable remain valid. Decreasing the acceptance window to accommodate the potential of an increase to pump runout flow, assures that the current limits on pump runout flows continue to be met. This change ensures pump integrity is maintained during the accident. The reduction of the flow by throttling valves to compensate for the potential suction boost remains within the current analyses and therefore more conservative values are being proposed. Additionally, the proposed change balances the pump flows more appropriately by differentiating between the hot and cold leg alignments. Flow to the reactor core is unaffected by the very slight reduction in the upper flow limits. Since the design limitations continue to be met and the integrity of the reactor coolant system pressure boundary is not challenged, offsite dose assumptions and calculations remain valid. Further, the ECCS is post-accident mitigation system and probability of an accident is not increased by this proposed change. Lastly, the correction of double use of the word ''the'' in Salem Unit 1 Technical Specification section 4.5.2.h.1.a is of editorial nature.

Based on the above information, the proposed changes do not increase the risk 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. No new single failures are initiated. The proposed changes will therefore not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change addresses suction boost by changing the Technical Specification surveillance acceptance criteria. The typographical correction is of editorial nature.

3. The proposed change does not involve a significant reduction in a margin of safety. The evaluation of LOCA accident analysis previously performed by Westinghouse continues to be met and verifies that, with the proposed changes to the TS, plant operations will be maintained within the bounds of safe, analyzed conditions as defined in the UFSAR [Updated Final Safety Analysis Report] and that conclusions presented in the UFSAR remain valid. The peak cladding temperatures (PTC) remains unchanged as no effective differences in the operating parameters have occurred. The typographical correction is of editorial nature. The proposed changes will therefore 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

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

Date of amendments request: March 7, 1997

Description of amendments request: The proposed amendments would allow operability testing for the containment isolation valves listed in Table 3.6-1 of the Technical Specifications during a defueled status. These proposed changes are technically consistent with the requirements of NUREG-1431, Revision 1, ''Westinghouse Standard Technical Specifications,'' issued on April 7, 1995.

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

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

The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated in the FSAR [Final Safety Analysis Report]. The proposed changes have no impact on the probability of an accident. The containment isolation valves will continue to require operability testing. Allowing the testing to be performed when the unit is in a defueled status will have no impact on any accidents previously evaluated. The net effect of these changes is not significant and, as a result, does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes to the Technical Specifications do not increase the possibility of a new or different kind of accident than any accident already evaluated in the FSAR. No new limiting single failure or accident scenario has been created or identified due to the proposed changes. Safety-related systems will continue to perform as designed. The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes do not involve a significant reduction in the margin of safety. Although, as a result of these proposed changes, the containment isolation valves could be tested for operability while the unit is in a defueled state, there is no impact in the accident analyses. These proposed changes are technically consistent with the requirements of NUREG-1431, Revision 1 which has already received the requisite review and approval of the NRC staff. Thus 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 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: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: August 22, 1996, as supplemented on March 28, 1997 (TS 96-02)

Description of amendment request: The proposed changes would revise Section 3.6.5 of the Sequoyah Technical Specifications (TS) and associated Bases to lower the minimum TS ice basket weight of 1,155 pounds to 1,071 pounds. This would reduce the overall weight of ice required in the ice condenser from 2,245,320 pounds to 2,082,024 pounds. The TVA license amendment request also proposed to extend the chemical analysis surveillance interval for the ice condenser ice bed from 12 months to 18 months based on the provisions of Generic Letter 93-05.

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

Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

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

TVA proposes to modify the SQN Unit 1 and Unit 2 TSs [Technical Specifications] to revise Surveillance Requirement (SR) 4.6.5.1.d to lower SQN's minimum TS basket weight from 1,155 pounds (lbs) to 1,071 lbs, thus lowering the overall ice condenser weight from 2,245,320 lbs to 2,082,024 lbs.

The ice condenser system is provided to absorb thermal energy release following a loss-of-coolant accident (LOCA) or high energy line break (HELB) and to limit the peak pressure inside containment. The current containment analysis for SQN is based on a minimum of 993 lbs of ice per basket evenly distributed throughout the ice condenser at the end of an 18-month refueling cycle. The revised containment analysis shows that for the predicted sublimation rate of 15 percent for 18 months, an average basket weight of 922 lbs at the end of the 18-month period would ensure containment design pressure is not exceeded.

Based on TVA's evaluation and the revised containment analysis, TVA considers the reduction of ice weight to be acceptable for satisfying the safety function of the ice condenser for an 18-month ice weighing interval. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

TVA is also proposing to extend the surveillance interval as it pertains to the ice bed chemical analysis. Based on test results, both at SQN and the industry, the average boron concentration and pH changes are minimal; therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Elimination of the temperature at which the pH of the ice bed is determined is an administrative change. Future testing will be accomplished in accordance with American Society for Testing and Materials Standards recommendations. Therefore, this change cannot increase the probability of an accident and the consequences of an accident will not increase.

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

TVA's request to lower the TS limit for ice weight at the start of the surveillance interval will not result in a new or different kind of accident from that previously analyzed in SQN's Final Safety Analysis

Report. SQN's ice condenser serves to limit the peak pressure inside containment following a LOCA. TVA has evaluated the revised containment pressure analysis for SQN (Enclosure 4, Westinghouse WCAP-12455, Revision 1) and determined that sufficient ice would be present at all times to keep the peak containment pressure below SQN's containment design pressure of 12 pounds per square inch gage (psig). Therefore, this change would not result in a new or different kind of accident from any previously analyzed.

The proposed reduced testing frequency of the chemical composition of the ice bed does not change the manner in which the plant is operated. Additionally, the ice condenser is a passive system that reacts to an accident, but does not support plant operation on a daily basis. The reduced testing frequency of the ice bed chemical composition does not generate any new accident precursors; therefore, the possibility of a new or different kind of accident from any previously analyzed is not created.

Elimination of the temperature at which the pH of the ice bed is determined is an administrative change. This change cannot create the possibility of a new or different kind of accident.

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

The ice condenser system is provided to absorb thermal energy release following a LOCA and to limit the peak pressure inside containment. The current ice condenser analysis for SQN is based on a minimum of 993 lbs of ice per basket. The revised containment analysis changes the minimum ice weight assumed in the analysis to 922 lbs per basket.

The revised containment analysis shows that using an average basket weight of 1,071 lbs and a sublimation allowance of 15 percent, all bays would have an average basket weight of 922 lbs at the end of the 18-month surveillance interval. The revised analysis utilizes new mass and energy releases (refer to Westinghouse WCAP-10325-P-A), which substantially delays ice-bed meltout and limits the initial containment peak pressure to approximately 7.15 psig during the blowdown phase. The ice-bed meltout delay allows the second containment pressure peak, which is driven mainly by the decay heat, to be limited to approximately 11.45 psig, which is below the containment design pressure of 12 psig.

Based on TVA's evaluation and the revised containment analysis, TVA considers the reduction of the average basket weight to be acceptable for satisfying the safety function of the ice condenser for the current 18-month interval. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

The proposal to extend the surveillance from 12 to 18 months does not change the boron concentration or pH requirements. Experience at Duke Power Company, as stated in NUREG-1366, indicates that these parameters do not change appreciably when verified every 9 months. SQN has a similar experience with a 12-month interval. Since the boron concentration and the post-LOCA pH requirements remain essentially the same, there is no reduction in the margin of safety.

Elimination of the temperature at which the pH of the ice bed is determined is an administrative change. Future testing will be accomplished in accordance with ASTM recommendations. The difference between the pH values determined at the current TS specified temperature and the temperature currently recommended by the ASTM standards is insignificant. Therefore, there is no 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: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402

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

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: September 12, 1996

Description of amendment request: The proposed change to the Technical Specifications is administrative in nature in that it would add the NRC standard fire protection license condition to each unit's Operating License and relocate the fire protection requirements from the Technical Specifications to the Updated Final Safety Analysis Report (UFSAR).

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

Specifically, operation of Surry Power Station with the proposed amendment will not:

1. Involve a significant increase in either the probability of occurrence or consequences of any accident or equipment malfunction scenario that is important to safety and which has been previously evaluated in the UFSAR. The requirements of the Fire Protection Program have not been changed by theproposed amendment. Relocation of the Fire Protection Program requirements into the UFSAR and station procedures does not decrease any portion of the program. The same fire protection requirements exist as before the change.

2. Create the possibility of a new or different type of accident than those previously evaluated in the safety analysis report. The requirements of the Fire Protection Program have not been changed by the proposed amendment. This is an administrative change to relocate the Fire Protection Program requirements from the Technical Specifications to the UFSAR and station procedures. Consequently, the possibility of a new or different kind of accident from any accident previously evaluated has not been created.

3. Involve a significant reduction in a margin of safety. Implementation of the Fire Protection Program requirements is assured by the UFSAR and station procedures. Since the rogram is being retained intact, there is no 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: Swem Library, College of William and Mary, Williamsburg, Virginia 23185.

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

NRC Project Director: Mark Reinhart, Acting

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

Date of amendment request: January 16, 1997

Description of amendment request: The proposed amendments (Point Beach Nuclear Plant (PBNP) Technical Specifications (TS) Change Request (TSCR) 191) would revise the minimum boron concentration required in the refueling water storage tank(s)(RWST), boric acid storage tanks (BAST), and safety injection (SI) accumulators during normal operation; the minimum boron concentration of primary coolant during refueling conditions; and the minimum boron concentration in the reactor when positive reactivity could be added and/or boron dilution could occur and containment integrity is not intact. These changes are necessary to accommodate the planned extension of the operating cycle from 12 months to 18 months. The licensee proposes to change TS 15.3.2, ''Chemical and Volume Control System,'' TS 15.3.3, ''Safety Injection and Residual Heat Removal Systems,'' TS 15.3.6, ''Containment System,'' TS 15.3.8, ''Refueling,'' and associated Bases.

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

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

The probabilities of accidents previously evaluated are based on the probability of initiating events for these accidents. Initiating events for accidents previously evaluated are described in the PBNP FSAR [final safety analysis report].

In effect, the proposed changes will result in: (1) higher boron concentrations of primary coolant during refueling, and (2) higher boron inventories in the RWSTs, BASTs, and SI accumulators. These changes do not require hardware changes or changes to the operation of accident-mitigating equipment. These changes relate to the performance capability of particular accident mitigation systems; equipment that is not postulated to cause accidents. Therefore, these proposed changes do not cause an increase in the probabilities of any accidents previously evaluated.

The consequences of accidents previously evaluated in the PBNP FSAR are determined by the results of analyses that are based on initial conditions of the plant, the type of accident, transient response of the plant, and the operation and failure of equipment and systems.

In effect, the proposed changes will result in: (1) higher boron concentrations of primary coolant during refueling, and (2) higher boron inventories in the RWSTs, BASTs, and SI accumulators. These increased boron concentrations do not increase the probability that engineered safety features equipment will fail, nor do these changes affect the capability of this equipment to operate as required for the accidents previously evaluated in the PBNP FSAR. These changes do not require hardware changes or changes to the operation of accident-mitigating equipment.

The consequential effects of a lower containment spray pH will not affect the capability of the containment spray to remove elemental iodine during design basis LOCA [loss-of-coolant accident] accidents. Also, the consequential reduction in containment sump water pH will not affect the fluid's capability to retain elemental iodine, nor will it adversely increase the potential corrosion rates for materials inside containment if the sump water is sprayed into containment during the recirculation phase of a LOCA.

Another consequence of injecting a higher concentration boric acid solution into the core during a LOCA may be an abbreviated onset to boron precipitation in the post-LOCA core. An incremental change in the boron injection concentration would not have significant effect on the postulated onset, but each core reload safety evaluation will continue to verify that the existing emergency operating procedures accommodate the potential for boron precipitation.

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

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

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

The changes proposed by this TSCR do not create any new or different accident initiators or sequences because these changes to minimum boron concentrations will not cause failures of equipment or accident sequences different than the accidents previously analyzed. No new equipment interfaces are created, and no new materials or fluids are introduced. The incremental increase in boron concentrations will not create a failure mechanism not previously known and evaluated. Therefore, these proposed TS changes do not create the possibility of an accident of a different type than any previously evaluated in the PBNP FSAR.

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

The margins of safety for Point Beach are based on the design and operation of the reactor and containment and the safety systems that provide their protection. Plant safety margins are established through Limiting Conditions for Operation, Limiting Safety System Settings and Safety Limits specified in the Technical Specifications. The proposed Technical Specification changes to refueling water storage tank (RWST), SI accumulator, and BAST boron inventory requirements have all been evaluated to preserve the shutdown capability described in the associated bases (boration from just critical, hot zero or full power, peak xenon with control rods at the insertion limit, to xenon-free cold shutdown with the highest worth control rod assembly fully withdrawn). Similarly, the proposed TS change to the minimum boron concentration of the primary coolant system for refueling operations have been evaluated to preserve the subcriticality margin described in the associated TS bases (i.e., 5% [delta] k/k in the cold condition with all rods inserted).

Because there are no changes to any of these margins, the proposed license amendment does not involve a reduction in any margin of safety.

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

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

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

NRC Project Director: John N. Hannon

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

Date of amendment request: January 21, 1997

Description of amendment request: The proposed amendments (Point Beach Nuclear Plant (PBNP) Technical Specifications (TS) Change Request 195) would revise TS Section 15.6.11, ''Radiation Protection Program,'' to update all references to 10 CFR Part 20, ''Standards for Protection Against Radiation,'' to restore consistency between 10 CFR Part 20 regulations and the PBNP TS.

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

1. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not create a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendments are administrative in nature, providing consistency between the Point Beach licenses and Commission regulations. The amendments do not affect the operation or maintenance of any PBNP structure[,] system or component. In addition, the regulations and proposed changes provide more conservative determinations of high radiation areas, thereby potentially resulting in lower personnel radiation exposures during normal operation and post accident. The consequences of an accident related to personnel radiation exposures may be reduced.

2. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendments are administrative only and do not affect the operation or maintenance of any structure[,] system or component at Point Beach Nuclear Plant. No new systems or components are introduced. Therefore, no new accident initiators or sequences result from any previously evaluated.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not create a significant reduction in a margin of safety.

The proposed amendments are administrative and reflect regulatory requirements that are more conservative than those presently reflected in the PBNP Technical Specifications. These more conservative requirements result in more conservative designation of high radiation areas thereby providing additional margins of safety related to the control of radiation exposures to personnel. No structure[,] system or component at PBNP at PBNP is changed[,] thereby maintaining the margins of safety for the operation of the Point Beach Nuclear Plant.

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

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

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

NRC Project Director: John N. Hannon

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

Date of amendment request: January 24, 1997

Description of amendment request: The proposed amendments (Point Beach Nuclear Plant (PBNP) Technical Specifications (TS) Change Request (TSCR) 193) would revise TS 15.5.4, ''Fuel Storage,'' to increase fuel assembly enrichment limits to 5.0 w/o U-235 while maintaining Keff in the storage pools (spent fuel pool and new fuel storage racks) less than 0.95.

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

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

The proposed changes do not involve a change to structures, systems, or components which would affect the probability or consequences of an accident previously evaluated in the PBNP Final Safety Analysis Report (FSAR). The only relevant concern with respect to increasing enrichment limits in the spent fuel pool and new fuel storage racks is one of criticality. The proposed changes use the same criticality limit used in the current Technical Specifications. Therefore, margin to safe operation of Units 1 and 2 is maintained. The probability and consequences of an accident previously evaluated are dependent on this criticality limit. Because the limit will not change, the probability and consequences of those accidents previously evaluated will not change.

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

The proposed changes do not involve a change to plant design. The proposed increase in spent fuel pool and new fuel storage racks fuel assembly enrichment limits maintains the margin to safe operation of Units 1 and 2 because the criticality limit for the spent fuel pool and new fuel storage racks will not change. These changes do not affect any of the parameters or conditions that contribute to the initiation of any accidents. Because the criticality limit remains the same, these changes have no effect on plant operation, design, or initiation of any accidents. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes maintain the margin to safe operation of Units 1 and 2. The margin of safety is based on the criticality limit of the spent fuel pool and the new fuel storage racks. Because this limit will not change, the margin of safety will not be affected. Therefore, the proposed changes will not create 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: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241

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

NRC Project Director: John N. Hannon

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

Date of amendment request: February 12, 1997, as supplemented on March 11, 1997

Description of amendment request: The proposed amendments (Point Beach Nuclear Plant (PBNP) Technical Specifications (TS) Change Request 196) would relocate turbine overspeed protection specifications, limiting conditions for operation, surveillance requirements, and associated bases from TS Section 15.3.4, ''Steam and Power Conversion System,'' and Section 15.4.1, ''Operational Safety Review,'' to the Final Safety Analysis Report (FSAR) in accordance with Generic Letter 95-10, ''Relocation of Selected Technical Specifications Requirements Related to 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. Operation of Point Beach Nuclear Plant in accordance with the proposed amendments will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendments administratively relocate turbine overspeed protection Specifications to the Point Beach Final Safety Analysis Report (FSAR). The Specifications will be transferred verbatim, except for the turbine load limit with the crossover steam dump system inoperable, which has already been evaluated under 10 CFR 50.59 and will be conservatively reduced. In addition, the regulatory requirements of 10 CFR 50.55a, ''Codes and Standards, '' will still apply to the relocated Specifications. Therefore, operation of Point Beach Nuclear Plant in accordance with the proposed amendments cannot create an increase in the probability or consequences of an accident previously evaluated.

2. Operation of Point Beach Nuclear Plant in accordance with the proposed amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendments administratively relocate Specifications to the FSAR and in one case result in a more conservative operating limit. Therefore, operation of Point Beach Nuclear Plant in accordance with the proposed amendments cannot create a new or different kind of accident from any accident previously evaluated.

3. Operation of Point Beach Nuclear Plant in accordance with the proposed amendments will not create a significant reduction in a margin of safety.

The proposed changes are administrative in nature. There is no physical change to the facility, its systems, or its operation, except for the conservative reduction of the turbine load limit with the crossover steam dump system inoperable which has already been justified via 10 CFR 50.59. Therefore, operation of PBNP in accordance with the proposed amendments cannot result in 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: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241

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

NRC Project Director: John N. Hannon

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

Date of amendment request: February 17, 1997; supersedes March 24, 1995, as supplemented by letter dated August 16, 1995, amendment request.

Description of amendment request: This amendment request proposes to revise Technical Specification 1.7, ''Containment Integrity,'' Technical Specification 3/4.6.1, ''Containment Integrity,'' and Technical Specification 3/4.6.3, ''Containment Isolation Valves.'' These proposed changes would relocate Technical Specification Table 3.6-1, ''Containment Isolation Valves,'' to the Wolf Creek Generating Station (WCGS) procedures. This proposed change is in accordance with the guidance provided in Generic Letter 91-08, ''Removal of Component Lists from Technical Specifications,'' dated May 6, 1991. In addition, this request proposes that the August 16, 1996, supplemental submittal that provided an additional footnote allowing an increased outage time for certain component cooling water system valves be withdrawn. The determination that the additional footnote is not required supersedes the staff's proposed no significant hazards consideration determination evaluation for the requested changes that was published on September 27, 1995 (60 FR 49949).

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

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

The proposed changes simplify the technical specifications, meet the regulatory requirements for control of containment isolation, and are consistent with the guidelines of GL 91-08. The procedural details of Technical Specification Table 3.6-1 have not been changed, but only relocated to a different controlling document. 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 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 function. 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 Technical Specification Table 3.6-1 to appropriate plant procedures does not alter the basic regulatory requirements for containment isolation and will not adversely affect containment isolation capability for Coordinator credible accident scenarios. Adequate control of the content of the table is assured by existing plant procedures.

The proposed relocation of Technical Specification Table 3.6-1 does not alter current technical specification requirements for containment isolation valve operability. The LCO and Surveillance Requirements would be retained in the revised technical specifications. Therefore, the proposed change will not affect the meaning, application, and function of the current technical specification requirements for the valves in Table 3.6-1.

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

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

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

NRC Project Director: William H. Bateman

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

Date of amendment request: March 18, 1997

Description of amendment request: This license amendment request revises Technical Specification Surveillance Requirement 4.5.2.c to clarify when a containment entry visual inspection is required. This proposed change to reduce the visual inspection requirement to at least once daily is in accordance with the guidance provided in Generic Letter 93-05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation.''

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

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

Implementing the proposed change could potentially increase the chances of loose debris (trash, rags, clothing, etc.) being left in containment for some period of time greater than would be allowed under current surveillance requirements. However, the proposed change also clarifies that the visual inspection must be performed at least once daily. Therefore, the period of time that debris could be left uncontrolled inside containment would still be less than 24 hours. Based on work controls placed on material entry/exit into containment and personnel training on housekeeping controls inside containment, and the results of past surveillances, it is unlikely that a significant amount of debris would be left uncontrolled inside containment for this period of time. Also, based on containment sump design, relatively small amounts of debris would not be sufficient to cause a significant amount of blockage of the sump screens.

The probability of occurrence of a previously evaluated accident is not increased because the reduced frequency of the visual inspection does not cause a significant impact on the possibility of containment sump screen blockage. Therefore containment sump operability is not affected by the proposed change. In addition, the proposed change will not result in any changes to the design or operation of any plant systems or components.

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

The proposed change decreases the frequency of performing a visual inspection for loose debris in containment, but does not result in a change to the design or operation of any plant system or component. The purpose of the inspection is to ensure that there is no loose debris, left in containment following a containment entry, that could potentially block the containment sump screens during LOCA conditions. Delaying this inspection until the last containment entry each day will not result in a significant amount of debris being left in containment, based on housekeeping practices controlling the entry/removal of trash and material into/from containment; training of employees to increase awareness of material control in radiologically-controlled areas; and retaining the requirement to perform a visual inspection at least once per day when containment entries are made (during periods when containment integrity is established), thereby ensuring that trash and debris can be identified and removed on a daily basis (on days containment entries are made).

Based on the above, 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 purpose of performing a visual inspection of areas affected by a containment entry is to ensure any debris or trash generated by the activity during the containment entry is identified and removed from containment. This ensures that no trash or debris is left in containment that could be transported to and block the containment sump screens during LOCA conditions. Based on current material control and housekeeping practices imposed on containment entry/exit, and past inspection results, reducing the surveillance requirement to a once per day basis, on days containment entries are made, would not result in a significant amount of trash or debris being left in containment following completion of the entry, and any such material would be identified and removed prior to the end of the day. The amount of trash or debris that could be left in containment for a 24 hour period would be significantly less than the amount that would be required to cause sump screen blockage sufficient to affect sump performance. Therefore, the proposed change will not result in a significant reduction in the margin of safety of any plant system or equipment.

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

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

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

NRC Project Director: William H. Bateman

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

Date of amendment request: March 18, 1997

Description of amendment request: This license amendment request revises Technical Specification Section 5.3.1, Fuel Assemblies, to allow the use of an alternate zirconium based fuel cladding material, ZIRLO. Wolf Creek Nuclear Operating Corporation (WCNOC) is planning to insert Westinghouse fuel assemblies containing ZIRLO fuel rod cladding during the ninth refueling outage, which is currently scheduled to begin in late September 1997.

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 methodologies used in the accident analysis remain unchanged. The proposed changes do not change or alter the design assumptions for the systems or components used to mitigate the consequences of an accident. Use of ZIRLO fuel cladding does not adversely affect fuel performance or impact nuclear design methodology. Therefore accident analyses are not impacted.

The operating limits will not be changed and the analysis methods to demonstrate operation within the limits will remain in accordance with NRC approved methodologies. Other than the changes to the fuel assemblies, there are no physical changes to the plant associated with this technical specification change. A safety analysis will continue to be performed for each cycle to demonstrate compliance with all fuel safety design basis.

VANTAGE 5H with IFMs fuel assemblies with ZIRLO clad fuel rods meet the same fuel assembly and fuel rod design bases as other VANTAGE 5H with IFMs fuel assemblies. In addition, the 10 CFR 50.46 criteria are applied to the ZIRLO clad rods. The use of these fuel assemblies will not result in a change to the reload design and safety analysis limits. The clad material is similar in chemical composition and has similar physical and mechanical properties as Zircaloy-4. Thus, the cladding integrity is maintained and the structural integrity of the fuel assembly is not affected. ZIRLO cladding improves corrosion performance and dimensional stability. No concerns have been identified with respect to the use of an assembly containing a combination of Zircaloy-4 and ZIRLO clad fuel rods. Since the dose predictions in the safety analyses are not sensitive to fuel rod cladding material, the radiological consequences of accidents previously evaluated in the safety analysis remain valid.

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

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

VANTAGE 5H with IFMs fuel assemblies with ZIRLO clad fuel rods satisfy the same design bases as those used for other VANTAGE 5H with IFMs fuel assemblies. All design and performance criteria continue to be met and no new failure mechanisms have been identified. Since the original design criteria are met, the ZIRLO clad fuel rods will not be an initiator for any new accident or malfunction of equipment important to safety. The ZIRLO cladding material offers improved corrosion resistance and structural integrity.

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

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

Use of ZIRLO cladding material does not change the VANTAGE 5H with IFMs reload design and safety limits. The use of these fuel assemblies will take into consideration the normal core operating conditions allowed in the Technical Specifications. For each cycle reload core, the fuel assemblies will be evaluated using NRC approved reload design methods, including consideration of the core physics analysis peaking factors and core average linear heat rate effects.

The use of Zircaloy-4, ZIRLO or stainless steel filler rods in fuel assemblies will not involve a significant reduction in the margin of safety because analyses using NRC approved methodologies will be performed for each configuration to demonstrate continued operation within the limits that assure acceptable plant response to accidents and transients. These analyses will be performed using NRC approved methods that have been approved for application to the fuel configuration.

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

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

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

NRC Project Director: William H. Bateman

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.

Carolina Power & Light Company, et al., Docket Nos. 50-325 and 50-324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of amendments request: March 27, 1997

Description of amendments request: The proposed amendments would revise the Technical Specifications for the Brunswick Steam Electric Plant Units 1 and 2 to eliminate certain instrumentation response time testing requirements in accordance with NRC-approved BWR Owners Group Topical Report NEDO-32291-A, ''System Analysis for the Elimination of Selected Response Time Testing Requirements.''Date of publication of individual notice in **Federal Register:** April 1, 1997 (62 FR 15542)

Expiration date of individual notice: May 1, 1997

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

Southern Nuclear Operating Company, Inc., Docket No. 50-348, Joseph M. Farley Nuclear Plant, Unit No. 1, Houston County, Alabama

Date of amendment request: March 25, 1997

Description of amendment request: The proposed amendment would modify Technical Specification 3/4.4.9, ''Specific Activity,'' and associated Bases to reduce the limit associated with dose equivalent iodine-131. The steady-state dose equivalent iodine-131 limit would be reduced by 40 percent from .5 [micro]Curie/gram to .3 [micro]Curie/gram.

Date of publication of individual notice in **Federal Register:** April 4, 1997 (62 FR 16201)

Expiration date of individual notice: May 5, 1997

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

Notice Of Issuance Of Amendments To Facility Operating Licenses

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

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

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

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

Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of application for amendments: February 19, 1997, as supplemented April 3, 1997.

Brief description of amendments: The amendments would delete the 24/48 Volt direct current (Vdc), batteries, battery chargers and distribution systems from the Technical Specifications (TSs) for Unit 3, by adding a footnote to indicate that these TSs are only applicable to Unit 2. All safety-related loads associated with the 24/48 Vdc batteries for Unit 3 will be relocated to other safety-related battery systems which are in the TSs.

Date of issuance: April 10, 1997

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

Amendment Nos.: 156 and 151

Facility Operating License Nos. DPR-19 and DPR-25: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 5, 1997 (62 FR 10088). The April 3, 1997, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 10, 1997. No significant hazards consideration comments received: No

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

Consolidated Edison Company of New York, Docket No. 50-247, Indian PointNuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: August 14, 1996, as supplemented September 13, 1996

Brief description of amendment: The amendment revises Technical Specification Sections 3.3 and 6.9.1.9; and the basis of Section 3.3, 3.6 and 3.10. The changes incorporate the best estimate approach into the licensing basis for the Indian Point Unit No. 2 loss-of-coolant accident analysis.

Date of issuance: March 31, 1997

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

Amendment No.: 188

Facility Operating License No. DPR-26: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 29, 1997 (62 FR 4344) The September 13, 1996, supplemental letter did not change the initial proposed no significant hazards consideration.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 31, 1997.No significant hazards consideration comments received: No

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

Consolidated Edison Company of New York, Docket No. 50-247, Indian PointNuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: February 14, 1997, as supplemented March 12, 1997.

Brief description of amendment: The amendment revises Technical Specification Section 4.13-2 to allow a one-time extension of the interval for steam generator tube inspection. Specifically, the date for commencement of the steam generator tube inspection is extended from April 14, 1997 to May 2, 1997.

Date of issuance: April 9, 1997

Effective date: As of the date of issuance to be implemented by April 14, 1997.

Amendment No.: 189

Facility Operating License No. DPR-26: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 4, 1997 (62 FR 9816) The March 12, 1997, supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 9, 1997.No significant hazards consideration comments received: No

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

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

Date of application for amendment: November 7, 1996

Brief description of amendment: The amendment revised Technical Specification 4.2.9, Service and Instrument Air System, to add an additional air compressor.

Date of issuance: April 2, 1997

Effective date: Effective the date of issuance.

Amendment No.: 118

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

Date of initial notice in **Federal Register:** December 18, 1996 (61 FR 66706) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 2, 1997.No significant hazards consideration comments received: No.

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

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: January 3, 1997, as supplemented by letter dated March 20, 1997

Brief description of amendments: The amendments revise Technical Specification Tables 3.3-2, 3.3-4, 3.3-5, 4.3-2 and Bases Sections 3/4.3.1 and 3/4.3.2 to eliminate the safety injection signal on low steam line pressure.

Date of issuance: April 3, 1997

Effective date: For Unit 1, as of the date of issuance to be implemented before startup from the next refueling outage; For Unit 2, as of the date of issuance to be implemented before startup from the current refueling outage

Amendment Nos.: 158 and 150

Facility Operating License Nos. NPF-35 and NPF-52: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 29, 1997 (62 FR 4345) The March 20, 1997, letter provided clarifying information that did not change the scope of the original January 3, 1997, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 3, 1997.No significant hazards consideration comments received: No

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

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 application for amendments: February 5, 1997

Brief description of amendments: The amendments reflect replacement of the existing source and intermediate range nuclear instrumentation with a new source range and wide range nuclear instrumentation system that provides more channels and continuous coverage from the Source Range to above the Power Range.

Date of issuance: March 31, 1997

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

Amendment Nos.: 223, 223, 220

Facility Operating License Nos. DPR-38, DPR-47, and DPR-55: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 26, 1997 (62 FR 8796) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 31, 1997.No significant hazards consideration comments received:

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691

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

Date of amendment request: October 16, 1996

Brief description of amendment: The amendment changes the Appendix A Technical Specifications by revising Table 4.3-1 to expand the applicability for Core Protection Calculator (CPC) operability and to allow the use of a cycle independent shape annealing matrix in the CPCs.

Date of issuance: April 11, 1997

Effective date: April 11, 1997, to be implemented within 60 days

Amendment No.: 125

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

Date of initial notice in **Federal Register:** February 12, 1997 (62 FR 6575) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 11, 1997No significant hazards consideration comments received: No.

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

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

Date of amendment request: December 2, 1996 as supplemented by letter dated February 4 and March 14, 1997

Brief description of amendment: The amendment changes the Technical Specifications to reflect the approval for the licensee to use of the new Containment Leakage Rate Testing Program as required by 10 CFR Part 50 Appendix J, Option B for Waterford Steam Electric Station, Unit 3.

Date of issuance: April 10, 1997

Effective date: April 10, 1997

Amendment No.: 124

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

Date of initial notice in **Federal Register:** January 15, 1997 (62 FR 2189) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 10, 1997.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, et al., Docket No. 50-335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of application for amendment: December 9, 1996

Brief description of amendment: This amendment modifies technical specifications for selected cycle-specific reactor physics parameters to refer to the St. Lucie Unit 1 Core Operating Limits Report for limiting values.

Date of issuance: April 1, 1997

Date of issuance: April 1, 1997

Effective date: April 1, 1997

Amendment No.: 150

Facility Operating License No. NPF-16: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 15, 1997 (62 FR 2189) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 1, 1997. No significant hazards consideration comments received: No.

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

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

Date of applications for amendment: June 20, 1995, as supplemented August 30, 1995, and January 17, 1996

Brief description of amendment: The amendment relocates the applicable requirements of Technical Specification (TS) 3.6.3 for the main steam line isolation valves (MSIVs) to TS 3.7.1.5, ''Main Steam Line Isolation Valves.'' In addition, the Applicability section of TS 3.7.1.5 is revised to indicate that Specification 3.7.1.5 is applicable in Mode 1 and in Modes 2, 3, and 4, except where all MSIVs are closed and deactivated (i.e., in Modes 2, 3, and 4, TS 3.7.1.5 is applicable only if the MSIVs are open). Also, the Action Statement for the Limiting Condition for Operation 3.7.1.5 has been revised using the guidance of the Improved Standard Technical Specifications for Westinghouse plants (NUREG-1431). The amendment also deletes a license requirement to submit responses to and to implement requirements of Generic Letter 83-28, because the requirement has been completed. Generic Letter 83-28 pertains to the Salem anticipated transient without scram event. In addition, the amendment incorporates TS Bases submitted by Northeast Nuclear Energy Company by letters dated June 20, 1995, February 5, 1996, and March 21 and 26, 1997. Since all four Bases changes affect Section B 3/4.7 of the TS, the NRC staff is using them in a group to avoid errors in revising the TS.

Date of issuance: April 10, 1997

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

Amendment No.: 136

Facility Operating License No. NPF-49: Amendment revised the Technical Specifications and License Condition.

Date of initial notice in **Federal Register:** August 2, 1995 (61 FR 39445) and February 28, 1996 (61 FR 7555)The August 30, 1995, letter provided clarifying information that did not change the scope of the June 20, 1995, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 10, 1997.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 06385

Northeast Nuclear Energy Company, et al., Docket Nos. 50-245, 50-336, and 50-423, Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3, New London, Connecticut

Date of application for amendments: February 3, 1997

Brief description of amendments: The amendments revise Section 6, ''Administrative Controls,'' of the Millstone Unit Nos. 1, 2, and 3 Technical Specifications to reflect organizational changes that have been implemented in the Nuclear Division.

Date of issuance: April 10, 1997

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

Amendment Nos.: 99, 206, and 135

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

Date of initial notice in **Federal Register:** February 26, 1997 (62 FR 8800) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 10, 1997.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 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut 06385

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

Date of application for amendments: November 18, 1996

Brief description of amendments: These amendments change the Technical Specifications for Susquehanna Steam Electric Station (SSES), Units 1 and 2 by increasing the maximum isolation times for reactor core isolation cooling inboard warm-up line isolation valves from 3 seconds to 12 seconds, high pressure core injection inboard warm-up line siolation valves from 3 seconds to 6 seconds and reactor recirculation process sample line isolation valves from 2 seconds to 9 seconds.

Date of issuance: April 7, 1997

Effective date: Both units, as of date of issuance, to be implemented within 30 days.

Amendment Nos.: 164 and 135

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

Date of initial notice in **Federal Register:** January 15, 1997 (61 FR 2191) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 7, 1997. No significant hazards consideration comments received: No

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

Pennsylvania Power and Light Company, Docket No. 50-388, Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania

Date of application for amendment: March 17, 1997

Brief description of amendment: The amendment modifies the Design Features Section 5.3.1 of the Technical Specifications to reflect the Atrium-10 design and would include a Siemens Power Corporation topical report in Section 6.9.3.2 to reflect mechanical design criteria for this fuel. This change would allow this fuel to be loaded into the core only under Operational Condition 5 (refueling) and does not permit startup or power operation using the Atrium-10 fuel.

Date of issuance: April 9, 1997

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

Amendment No.: 136

Facility Operating License No. NPF-22: This amendment revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: Yes (62 FR 14167) March 25, 1997. 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 for an opportunity to request a hearing by April 24, 1997, 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 determination of no significant hazards consideration are contained in a Safety Evaluation dated April 9, 1997.

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

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

Southern Nuclear Operating Company, Inc., 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: September 19, 1996, as supplemented December 17, 1996, January 23 and 31, March 21 and April 4, 1997

Brief description of amendments: The amendments revise the surveillance requirements addressing the reactor vessel pressure and temperature limits.

Date of issuance: April 4, 1997

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

Amendment Nos.: 206 and 147

Facility Operating

Local Public- Document -Room locations: ments revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 2, 1997 (62 FR 128) The December 17, 1996, January 23 and 31, March 21, 1997, and April 4, 1997, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 4, 1997.No significant hazards consideration comments received: No

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

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

Date of application for amendments: October 18, 1996 as supplemented March 12, March 17, April 4, and April 9, 1997 (TS 96-05)

Brief description of amendments: The amendments change the Technical Specifications (TS) by revising TS 3/4.4.5 and 3.4.6.2 and associated Bases to permanently incorporate requirements for steam generator tube inspections and repair in the Sequoyah Nuclear Plant, Units 1 and 2 TS.

Date of issuance: April 9, 1997

Effective date: As of the date of issuance to be implemented no later than 45 days of its issuance.

Amendment Nos.: 222 and 213

Facility Operating License Nos. DPR-77 and DPR-79: Amendments revise the technical specifications and license conditions.

Date of initial notice in **Federal Register:** February 11, 1997 (62 FR 6276) The March 12, March 17, April 4, and April 9, 1997, letters provided clarifying information that did not change the scope of the October 18, 1996, application and the initial proposed no significant hazards consideration determination.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 9, 1997.No significant hazards consideration comments received: None

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

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

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

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

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

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

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

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

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By May 23, 1997, 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. 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).

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

Date of application for amendment: April 1, 1997

Brief description of amendment: The amendment revises Technical Specification Table 3.3-3 to correct administrative errors associated with the start logic of the turbine driven auxiliary feedwater pump.

Date of issuance: April 2, 1997

Effective date: April 2, 1997

Amendment No.: 119

Facility Operating License No. NPF-30: The amendment revised the Technical Specifications.Public comments requested as to proposed no significant hazards consideration: No.The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated April 2, 1997.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 200379

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

NRC Project Director: William H. Bateman

Dated at Rockville, Maryland, this 16th day of April, 1997.

For the Nuclear Regulatory Commission

Jack W. Roe,

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

[Doc. 97–10334 Filed 4–22–97; 8:45 am]

BILLING CODE 7590–01–F

SECURITIES AND EXCHANGE COMMISSION

Issuer Delisting; Notice of Application to Withdraw From Listing and Registration; (Donnelly Corporation, Class A Common Stock, $0.10 Par Value) File No. 1–9716

April 17, 1997.

Donnelly Corporation (''Company'') has filed an application with the Securities and Exchange Commission (''Commission''), pursuant to Section 12(d) of the Securities Exchange Act of 1934 (''Act'') and Rule 12d2–2(d) promulgated thereunder, to withdraw the above specified security (''Security'') from listing and registration on the American Stock Exchange, Inc. (''Amex'' or ''Exchange'').

The reasons cited in the application for withdrawing the Security from listing and registration include the following:

The Company has complied with Rule 18 of the Amex by filing with such Exchange a certified copy of preambles and resolutions adopted by the Company's Board of Directors authorizing the withdrawal of its common stock from listing on the Amex and by setting forth in detail to such Exchange the reasons for such proposed withdrawal, and the facts in support thereof. The Company became listed for trading on the New York Stock Exchange, Inc. (''NYSE'') pursuant to a Registration Statement on Form 8–A effective March 6, 1997.

In making the decision to withdraw its common stock from listing on the Amex, the Company considered: (a) that the Company believes that the NYSE will offer the Company's shareholders more liquidity over time than is presently available on the Amex; (b) that the Company believes that listing on the NYSE will offer greater visibility for the Company and its stock potential for greater institutional ownership; (c) that as the Company becomes an increasingly international company, it believes there will be advantages to having its stock listed on the NYSE, and (d) many of the companies which it regards as peers or leaders in its industry are listed on the NYSE.

Any interested person may, on or before May 8, 1997, submit by letter to the Secretary of the Securities and Exchange Commission, 450 Fifth Street, NW., Washington, DC 20549, facts bearing upon whether the application has been made in accordance with the rules of the exchanges and what terms, if any, should be imposed by the Commission for the protection of investors. The Commission, based on