September 2, 1994

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

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

Dear Mr. Chairman:

Public Law 97-415, enacted on January 4, 1983, amended Section 189 of the Atomic Energy Act of 1954 to authorize the Nuclear Regulatory Commission to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing.

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

Enclosed for your information is a copy of the Commission's Biweekly Notice of Applications and Amendments to Operating Licenses involving no significant hazards considerations, which was published in the Federal Register on August 17, 1994, on page 42332.

Sincerely,

ORIGINAL SIGNED BY

William T. Russell, Director
Office of Nuclear Reactor Regulation

Enclosure: Federal Register
Notice

cc: Senator Alan K. Simpson

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Wednesday
August 17, 1994

Part II

Nuclear Regulatory Commission

Operating Licenses, Amendments; No Significant Hazards Considerations; Biweekly Notices

UNITED STATES NUCLEAR REGULATORY COMMISSION

Biweekly Notice

Applications and Amendments to Facility Operating LicensesInvolving 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 July 25, 1994, through August 5, 1994. The last biweekly notice was published on August 3, 1994.

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

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

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

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

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

By September 16, 1994, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555 and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

contention will not be permitted to participate as a party.

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

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

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

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

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

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

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

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

Date of amendment requests: May 4, 1994

Description of amendment requests: These amendment requests would revise Limiting Condition for Operation (LCO) 3.4.8.3 and Surveillance Requirement 4.4.8.3.1, "Overpressure Protection Systems." Specifically, the LCO and surveillance requirements are revised to clarify that both shutdown cooling system (SCS) suction line relief valves shall be OPERABLE and aligned to provide overpressure protection not only during reactor (RCS) cooldown or heatup evolutions, but also during any steady state temperature periods maintained in the course of RCS cooldown or heatup evolutions.

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

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

The proposed amendments provide further clarification of the Technical Specifications and represent an additional operating limitation. Incorporating the noted clarification will not change the bases or assumptions contained in the safety analysis for this system. The most limiting low-temperature overpressure protection (LTOP) transients, the starting of an idle reactor coolant pump (RCP) and the inadvertent actuation of two high pressure safety injection (HPSI) pumps into a solid RCS, are not affected by the proposed clarification. Therefore, the proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Clarifying the applicability of the LCO's and surveillance for steady state periods achieved and maintained during either a heatup or cooldown evolution does not modify the design or operation of plant equipment. No new or different failure modes will be introduced by incorporating this clarification into the LCO and surveillance requirement. Therefore, the proposed amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The clarification will enhance LCO 3.4.8.3 and Surveillance Requirement 4.4.8.3.1 for heatup and cooldown evolutions by ensuring operators are aware of this applicability during periods of steady state conditions. This clarification does not involve a change to safety limits, setpoints, or design margins. As such, the proposed amendments will not involve a significant reduction in a margin of safety at PVNGS.

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

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

Attorney for licensees: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072-3999

NRC Project Director: Theodore R. Quay

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

Date of amendment requests: June 17 1994

Description of amendment requests: The proposed amendments would increase the minimum nitrogen accumulator pressure for the atmospheric dump valves (ADVs), as stated in the surveillance requirements of Technical Specification (TS) 3/4.7.1.6. The change to the Bases increases the minimum time the ADV accumulators must be operable.

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

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

The proposed Technical Specification change in the nitrogen accumulator supply minimum pressure will not increase the probability or consequences of any accident previously analyzed. The nitrogen accumulator pressure is normally maintained between 650-680 psig. Nitrogen pressure from the accumulator is reduced to 105 psig

prior to use in the operation of the ADVs. The pressure reduction will remain the same with the higher minimum accumulator pressure.

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

Increasing the nitrogen accumulator minimum pressure does not create any new or different accidents than those previously evaluated. The normal air supply (the Instrument Air System) to the ADV is maintained between 105 to 125 psig. Currently, nitrogen from the accumulator is reduced to 105 psig prior to use in the ADV. The increased minimum pressure in the accumulator will still be reduced to 105 psig prior to use in the ADV.

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

The limitation on maintaining the nitrogen accumulator at a certain pressure is to ensure that a sufficient volume of nitrogen is in the accumulator to operate the associated ADV. Maintaining a higher minimum pressure ensures that sufficient nitrogen will be available to maintain the unit at HOT STANDBY for four hours and an additional 9.3 hours to reach COLD SHUTDOWN under natural circulation conditions in the event of failure of the normal control air system. Therefore, the proposed change does not involve a reduction in a margin of safety.

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

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

Attorney for licensees: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072-3999

NRC Project Director: Theodore R. Quay

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

Date of amendment requests: July 12, 1994

Description of amendment requests: The proposed amendment would enhance the PVNGS Technical Specifications (TS) by adding a limiting condition for operation (LCO) action statement to Entry VIII B of Table 3.3-3, "Engineered Safety Features Actuation System Instrumentation." The proposed action statement would enhance safe plant operation by requiring timely plant shutdown if more than one of the new solid state degraded voltage relays in either train of 4.16kV are inoperable or not energized.

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

Standard 1--Involve a significant increase in the probability or consequence of an accident previously evaluated:

The proposed amendment will add an action statement to TS Table 3.3-3 entry VIII B which would allow eight hours to effect repairs. This action statement would be entered if more than one of the required four degraded voltage relays on either 4.16 kv bus is inoperable or not energized. If the eight hour allowed outage time is not met, the unit is placed in Hot Standby within six hours and in Cold Shutdown within the next thirty hours. Technical Specification 3.8.3.1 currently allows eight hours to restore a 4.16 kv bus in the event of a loss of power to that bus. The loss of degraded voltage relays on that bus does not impact plant nuclear safety any more than the loss of the bus itself. Furthermore, even with the loss of all four degraded voltage relays monitoring one 4.16 kv bus (for example, due to a blown 125 vdc circuit fuse), the loss-of-voltage relays on that bus, and the degraded voltage relays, as well as the loss-of-voltage relays monitoring the other bus would be unaffected. None of the UFSAR chapter 15 accident analyses are affected by this proposed amendment. The existing TS requirements and those components to which they apply are not altered by this TS amendment. There are no changes to the maintenance, surveillance, and/or qualification of any component/function in Table 3.3-3. Therefore, the addition of this proposed eight hour action statement to Table 3.3-3 entry VIII B does not increase the probability of occurrence or the consequences of any previously evaluated accident.

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

The TS requirements and the components to which they apply are not altered by this amendment. The new solid state degraded voltage relays in each 4.16 kv bus were installed under the 10 CFR 50.59 change process. APS [Arizona Public Service Company] determined that the installation created no unreviewed safety question. This amendment has no impact on plant maintenance, testing, shutdown equipment, or component qualification. Plant operational safety is enhanced by this amendment. Therefore, the possibility of a new or different kind of accident is not created by this amendment.

Standard 3--Involve a significant reduction in a margin of safety:

The TS does not alter existing TS requirements or those components to which they apply. More specifically, there is no impact on safe plant shutdown, maintenance, containment isolation capability, containment leakage rate, or the operability of safety related valves. Therefore, the addition of the proposed action statement to the TS will not involve reduction in a margin of safety for fission product release to the atmosphere.

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

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

Basis for proposed no significant hazards consideration determination:

Attorney for licensees: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072-3999

NRC Project Director: Theodore R. Quay

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

Date of amendments request: June 8, 1994

Description of amendments request: The proposed amendment would revise the Calvert Cliffs Nuclear Power Plant (CCNPP) Units 1 and 2 Technical Specification (TS) Section 4.7.1.2.c to extend the interval for three Auxiliary Feedwater (AFW) surveillance requirements from 18 to 24 months. Specifically, TS Section 4.7.2.c.1 requires the verification of each automatic valve in the flowpath actuate to its correct position and each AFW pump automatically start upon receipt of each AFW actuation system test signal; TS Section 4.7.2.c.2 requires verification that the AFW system is capable of providing a minimum 300 gallons per minute nominal flow to each leg. This request is one of a series of proposed license amendments that would eliminate the need for mid-cycle surveillance outages by extending 18-month frequency surveillances to every refueling outage (nominally each 24 months).

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

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

The Auxiliary Feedwater (AFW) System provides a safety-related source of feedwater to the steam generators to mitigate design

basis accidents involving loss of Main Feedwater. Failure of the AFW System is not an initiator for any previously analyzed accident. Therefore, the proposed change does not involve an increase in the probability of an accident previously evaluated.

A historical review of surveillance test results and system performance indicates that the AFW System is very reliable. In addition, monthly surveillances of the AFW System will continue to verify proper pump and valve operation. The AFW System reliability and monthly surveillances provide assurance that undetected system degradation will not occur between 24-month surveillances. Therefore, the AFW System will continue to perform its safety function and there will be no significant increase in the consequences of accidents. Therefore, the proposed Technical Specification changes do not increase the probability or consequences of an accident previously evaluated.

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

This requested revision to increase the interval for some AFW surveillances from 18' to 24 months does not involve a significant change in the design or operation of the plant. No hardware is being added to the plant as part of the proposed change. The proposed change will not introduce any new accident initiators. Therefore, this change would not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Does operation of the facility in accordance with the proposed amendment involve a significant reduction in a margin of safety?

The AFW System provides a margin of safety by providing a safety-related alternate supply of feedwater to the steam generator for removal of decay heat and cooldown of the Reactor Coolant System. The proposed changes do not affect the operation or design of the AFW System. Monthly surveillances and historical data provide assurance that the reduction in surveillance frequency will not adversely affect our ability to detect degradation in the system. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Michael L. Boyle

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

Date of amendment request: April 29, 1994

Description of amendment request: This amendment is an additional followup to the amendment request of May 29, 1992, published in the **Federal Register** on July 8, 1992 (57FR30242), which changed the Technical Specifications Section 1.0, Definitions, to accommodate a 24-month fuel cycle and which proposed the extension of the test intervals for specific surveillance tests. This amendment proposes extending the surveillance intervals to 24 months for the following additional surveillance tests:(1) Calibrate and test channels for Auxiliary Feedwater (AFW) initiation on steam generator water level (low-low).(2) Test channels for Auxiliary Feedwater initiation on trip of main feedwater pumps.The licensee's amendment proposal of November 25, 1992, requested approval for extending the surveillance interval of the Auxiliary Feedwater System to accommodate a 24-month fuel cycle and the approved change was issued in License Amendment No. 166. Subsequently, the licensee determined that two additional surveillances associated with this system had not been identified in the November 25, 1992, request. This amendment proposal requests approval of the additional surveillances. The changes requested by the licensee are in accordance with Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle."

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 change does not involve a significant hazards consideration since:

1. There is no significant increase in the probability or consequences of an accident.

The test results over the last four refuelings confirmed system operability with only one failure. This failure would not have impaired the ability of the auxiliary feedwater system to perform its intended safety function. The auxiliary feedwater system is redundant and diverse. The failure in the turbine driven pump did not impact the motor driven pumps.

Based on the historical test data, it is concluded that no significant increase in the probability or consequences of an accident would be incurred by extending the operating cycle due to an increased surveillance interval.

2. The possibility of a new or different kind of accident from any previously analyzed has not been created.

The failure noted from the past test data appears random in nature and would not have defeated the redundancy in design that exists in the AFW system. The AFW system would have been capable of performing its intended safety function and therefore a new or different kind of accident would not have been created.

3. There has been no reduction in the margin of safety.

Past historical data demonstrates that the AFW systems would perform their safety function for an extended operating cycle should the surveillance period be extended by several months.

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

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

Attorney for licensee: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Project Director: Pao Tsin Kuo

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

Date of amendment request: June 16, 1994

Description of amendment request: The proposed amendment would revise License Condition 2.K of the license issued August 24, 1981, to provide for compliance with the NRC-approved fire protection program as described in the Updated Final Safety Analysis Report and for making changes to the NRC-approved fire protection program; would delete fire protection Technical Specification (TS) Sections 3.13 and 4.14 which contain limiting conditions for operation and surveillance requirements, respectively, for the high-pressure water fire protection system, fire protection spray systems, penetration fire barriers, fire detection systems, fire hose stations and hydrants, and the cable spreading room halon system; would delete Section 6.2.2.f which contains fire brigade staffing requirements; would delete Section 6.4.2 which contains fire brigade training requirements; would add Section 6.5.1.6.1 to add fire protection program responsibilities to the Station Nuclear Committee; would add Section 6.8.1.e to require written procedures and administrative policies for the fire protection program; would delete

Section 6.9.2.b which requires a Special Report for inoperable fire protection and detection equipment; and would make corresponding changes to the Table of Contents and List of Tables.

Generic Letter (GL) 86-10, dated April 24, 1986, and GL 88-12, dated August 2, 1988, from the NRC provided guidance to licensees to request removal of the fire protection TS. The licensee's proposed amendment is in response to these GLs.

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 Commission has provided guidance concerning the application of the standards for determining whether a "Significant Hazards Consideration" exists by providing certain examples in 51 FR 7744 (dated March 6, 1986). Example (vii) of those involving no significant hazards considerations relates to "a change to conform a license to changes in the regulations, where the license change results in very minor changes to facility operations clearly in keeping with the regulations."

In this case, NRC Generic Letters 86-10 and 88-12, although not regulations, provide pertinent guidance relative to the above described proposed changes and implementation of the NRC fire protection regulations of 10 CFR 50.48(a). Specifically, the generic letters allow licensees to delete fire protection related technical specifications, provided that administrative requirements are added to technical specifications and a license condition is provided that requires the implementation and maintenance in effect of the approved fire protection program. Further, the generic letters provide for inclusion of the fire protection program into the UFSAR [Updated Final Safety Analysis Report] and permits future changes to the fire protection program without prior NRC approval, all as provided by the license condition and in accordance with the provisions of 10 CFR 50.59. Therefore, since the actions required by the generic letters have been taken and conform the license to the current interpretation of NRC fire protection regulations as described in Generic Letters 86-10 and 88-12, with no changes to facility operations, these proposed changes are in accordance with Example (vii) above.

In accordance with the requirements of 10 CFR 50.92, the proposed changes to the Technical Specifications are deemed not to involve any "Significant Hazards Considerations" because operation of Indian Point Unit No. 2 in accordance with these changes would not:

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

The fire protection program requirements are not affected in that the function, operation or surveillance requirements for any fire protection system or component are not being altered. The proposed changes simply relocate these requirements from the Technical Specifications to the UFSAR, are administrative in nature, and do not affect any other current plant equipment or practices. Therefore, the conclusions of current accident analyses are not affected. Further, as permitted by the proposed License Condition 2.K, changes in the NRC-approved fire protection program will require an evaluation per the criteria of 10 CFR 50.59 to determine that the proposed change will not involve an unreviewed safety question. Therefore, future changes to the fire protection program will be evaluated in accordance with appropriate criteria.

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

The proposed changes introduce no new mode of plant operation, do not involve physical modification to any structure, system or component, do not affect the function, operation or surveillance requirements for any equipment necessary for safe operation or shutdown of the plant or of fire protection equipment which protects such equipment, and do not involve any changes to setpoints or operating parameters. The changes are administrative only and all existing fire protection requirements are maintained. Therefore, the changes can not result in an unanalyzed accident. Further, as permitted by the proposed License Condition 2.K, changes in the NRC-approved fire protection program will require an evaluation per the criteria of 10 CFR 50.59 to determine that the proposed change will not involve an unreviewed safety question. Therefore, future changes to the fire protection program will be evaluated in accordance with appropriate criteria.

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

The existing fire protection program operability and surveillance requirements are retained as they are contained in the FPPP [Fire Protection Program Plan], and compliance will continue through proposed License Condition 2.K. Therefore, no margins of safety established by design or verified by testing to ensure operability of fire protection systems or components are affected. Further, as permitted by the proposed License Condition 2.K, changes in the NRC-approved fire protection program will require an evaluation per the criteria of 10 CFR 50.59 to determine that the proposed change will not involve an unreviewed safety question. Therefore, future changes to the fire protection program will be evaluated in accordance with appropriate criteria.

Based on the above discussion, since these proposed changes to the Indian Point Unit No. 2 Technical Specifications satisfy the criteria specified in 10 CFR 50.92, are similar to an example provided by the Commission of a change which involves "No Significant Hazards Considerations", and are not similar to any examples that involve a "Significant Hazards Consideration", Con Edison has determined that this amendment application does not involve any "Significant Hazards Considerations."

The proposed Technical Specification changes have been reviewed by the Station Nuclear Safety Committee and the Con Edison Nuclear Facilities Safety Committee. Both committees concur that these proposed changes do not represent any "Significant Hazards Considerations."

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

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

Attorney for licensee: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Project Director: Michael L. Boyle

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

Date of amendment request: July 8, 1994

Description of amendment request: The proposed amendment would revise Technical Specification Section 3.7, Auxiliary Electrical Systems to clarify offsite power availability requirements and to revise emergency diesel generator fuel oil availability requirements.

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

Consistent with the requirements of 10 CFR 50.92, the enclosed application involves no significant hazards based on the following information:

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

Response:

Neither the probability nor the consequence of an accident previously analyzed is increased due to the proposed changes. There are no changes on the existing offsite power supply configuration or on the existing diesel fuel oil supply system or inventory requirements. This proposed amendment will allow for three diesel operation when a fuel oil storage tank or transfer pump is unavailable. In the event of an accident at this time, the three diesel operation would allow for more than minimum safeguards to be available, with maximum safeguards available for the first part of the event.

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

Response:

The existing 138 kV and 13.8 kV offsite power reliability is maintained with this change. There is no impact on availability of the alternate AC system, the three gas turbines, with this change. This change is consistent with the original licensing basis that the AEC accepted for the diesel fuel oil supply system.

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

Response:

The proposed amendment does not involve a significant reduction in the margin of safety. The proposed amendment maintains the reliability of the preferred 138 kV and 13.8 kV offsite power and is consistent with the original licensing basis for diesel fuel oil inventory.

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

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

Attorney for licensee: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Project Director: Robert A. Capra

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

Date of amendment request: July 29, 1994

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) 3/4.4.5 and 3/4.4.6.2 and associated bases to allow the implementation of interim steam generator tube plugging criteria for the tube support elevations during cycle 11. The allowed primary-to-secondary operational leakage from any one steam generator is proposed to be reduced from 500 gallons per day (gpd) to 150 gpd. The total allowed primary-to-secondary operational leakage from all steam generators would be reduced from one gallon per minute (1440 gpd) to 450 gpd.

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?

Testing of model boiler specimens for free span tubing (no tube support plate (TSP) restraint) at room temperature conditions show[s] burst pressures in excess of 5000 psi for indications of outer diameter stress corrosion cracking with voltage measurement as high as 19 volts. Burst testing performed on intersections pulled from BVPS [Beaver Valley Power Station] with up to a 2.7 volt indication shows measured burst pressure in excess of 6600 psi at room temperature. Burst testing performed on pulled tubes from other plants with up to 7.5 volt indications show[s] burst pressures in excess of 6300 psi at room temperatures. Correcting for the effects of temperature on material properties and minimum strength levels (as the burst testing was done at room temperature), tube burst capability significantly exceeds the safety factor requirements of RG [Regulatory Guide] 1.121. As stated earlier, tube burst criteria are inherently satisfied during normal operating conditions due to the proximity of the TSP. Test data indicates that tube burst cannot occur within the TSP, even for tubes which have 100 percent through wall electric discharge machining (EDM) notches, 0.75 inch long, provided that the TSP is adjacent to the notched area. Since tube to TSP proximity precludes tube burst during normal operating conditions, use of the criteria must retain tube integrity characteristics which maintain a margin of safety of 1.43 times the bounding faulted condition steam line break (SLB) pressure differential. As previously stated, the RG 1.121 criterion requiring maintenance of a safety factor of 1.43 times the SLB pressure differential on tube burst is satisfied by 7/8 inch diameter tubing with bobbin coil indications with signal amplitudes less than 8.82 volts regardless of the indicated depth measurement. The plugging criteria (resulting in a projected end-of-cycle [EOC] voltage) compares favorably with the 8.82 volt structural limit considering the extremely slow apparent voltage growth rate of indications at BVPS. Using the established methodology of RG 1.121, the structural limit is reduced by allowances for uncertainty and growth to develop a beginning-of-cycle (BOC) repair limit which should preclude indications at EOC conditions which exceed the structural limit. The non-destructive examination (NDE) uncertainty component is 20.5 percent and is based on the EPRI [Electric Power Research Institute] Alternate Repair Criteria (ARC). A bounding growth allowance of 40 percent will be applied. This value is conservative for BVPS Unit 1. The BOC maximum allowable repair limit should not permit the existence of EOC indications (when the 40 percent growth and 20.5 percent uncertainty allowances are applied) which exceed the 8.82 volt structural limit. By adding NDE uncertainty allowances and an allowance for crack growth to the repair limit, the structural limit can be validated. Therefore, the maximum allowable BOC repair limit (RL) based on the structural limit of 8.82 volts can be represented by the expression:

RL + (0.205 X RL) + (0.40 x RL) = 8.82 volts, or the maximum allowable BOC repair limit can be expressed as:

RL = 8.82 volt structural limit/1.605 = 5.5 volts.

It is reasonable that this repair limit (5.5 volts) could be applied for IPC [interim plugging criterion] implementation to repair bobbin indications greater than 1.0 or 2.0 volts independent of RPC [rotating pancake coil] confirmation of the indication. The analyses were performed based on a 1.0 or 2.0 volt repair limit. Duquesne Light Company has chosen to use a steam generator tube repair limit of 1.0 volt. Conservatively an upper limit of 3.6 volts will be used to assess tube integrity for those bobbin indications which are above 1.0 volt but do not have confirming RPC calls. This 3.6 volt upper limit for non-confirmed RPC calls is consistent with other recently approved IPC programs for the two other plants with 7/8 inch tubing that currently implement IPCs. Since the upper bound for repair of non-confirmed RPC is limited to a value far less than the limit associated with a full alternate criteria, the establishment of the repair limits are [is] judged to be independent of the pulled tube data base used.

The conservatism of the growth allowance used to develop the repair limit is shown by the most recent BVPS eddy current data. The average voltage growth for all indications was 16 percent while the average voltage growth for indications greater than 0.75 volts at BOC was 6 percent. The largest overall voltage growth in a particular steam generator was found in the "A" steam generator, which had an overall average growth of 25 percent. Only two tubes had an absolute voltage growth which exceeded 1.0 volt for Cycle 9. The maximum absolute voltage growth in the 1993 inspection was recorded to be 1.18 volts. Each of the last three inspections, which included 100 percent of all hot leg tubes, showed decreasing voltage growth trends in each successive inspection for all categories; overall voltage growth, growth of BOC indications less than 0.75 volts, and growth of indications greater than 0.75 volts. The decreasing voltage growth rate trend data indicates that DLC has good control of the ODSCC [outer diameter stress corrosion cracking] occurring in the BVPS Unit 1 steam generators and also implies that atypical voltage growth of a few indications is unlikely.

Relative to the expected leakage during accident condition loadings, it has been previously established that a postulated main SLB [steam line break] outside of containment but upstream of the main steam isolation valve (MSIV) represents the most limiting radiological condition relative to the IPC. In support of implementation of the interim plugging criteria, it will be determined whether the distribution of cracking indications at the TSP intersections at the end of Cycle 11 are [is] projected to be such that primary-to-secondary leakage would result in site boundary doses within a small fraction of the 10 CFR 100 guidelines. A separate calculation has determined this allowable SLB leakage limit to be 6.6 gpm in the faulted loop. This limit was calculated using the Technical Specification RCS [reactor coolant system] Iodine-131 activity level of 1.0 micro Curies per gram dose equivalent Iodine-131 and the recommended Iodine-131 transient spiking values consistent with NUREG-0800. The projected SLB leakage rate calculation methodology prescribed in Section 3.3 of draft NUREG-

1477 will be used to calculate EOC leakage. The log-logistic probability of leakage correlation will be used to establish the SLB leak rate used for comparison with the 6.6 gpm faulted loop allowable limit. Due to the relatively low voltage levels of indications at BVPS and low voltage growth rates, it is expected that the actual calculated leakage values will be far less than this limit. Additionally, the current Iodine-131 levels as of May 1994 at BVPS are about 1000 times less than the Technical Specification limit of 1.0.

Application of the criteria requires the projection of postulated SLB leakage, based on the projected EOC voltage distribution for the upcoming cycle. Projected EOC voltage distribution is developed using the most recent EOC eddy current results and a voltage measurement uncertainty. Data indicate that a threshold voltage of 2.8 volts would result in through wall cracks long enough to leak at steam line break conditions. Draft NUREG-1477 requires that all indications to which the IPC are applied must be included in the leakage projection. Tube pull results from another plant with 7/8 inch tubing with a substantial voltage growth data base have shown that tube wall degradation of greater than 40 percent through wall was readily detectable either by the bobbin or RPC probe. The tube with maximum through wall penetration of 56 percent (42 percent average) had a voltage of 2.02 volts. This indication also was the largest recorded bobbin voltage from the EOC eddy current data. Based on the BVPS pulled tube and industry pulled tube data supporting a lower threshold for SLB leakage of 2.8 volts, inclusion of all IPC intersections in the leakage model is quite conservative. The ODSCC occurring at BVPS has historically resulted in relatively low voltage levels and has exhibited decreasing voltage growth trends over the last three inspections. BVPS has not identified ODSCC as a contributor to operational leakage. The current leakage levels at BVPS are negligible (less than 1 gpd). In order to satisfy the requirements of draft NUREG-1477, EOC 10 eddy current data will be used to calculate the projected SLB leakage according to draft NUREG-1477 methodology. Leakage calculated using the recommended EPRI leakage correlation will also be provided. Duquesne Light Company is requesting that the NRC review and approve the EPRI SLB leakage calculation methodology. Sufficient justification is included to establish acceptability of the EPRI leakage correlation based on criteria provided by the NRC in the February 8, 1994, Industry/NRC working meeting on the voltage based criteria.

In order to assess the sensitivity of application of the voltage based criteria upon SLB leakage, the EOC 9 eddy current results were used to calculate postulated EOC 10 leakage using both the NUREG-1477 methodology and EPRI correlation assuming that a 1.0 or 2.0 volt plugging limit were implemented at the BOC 10.

Results indicate SLB leakage of 0.46 gpm and 0.044 gpm using the NUREG and EPRI methodologies with an assumed probability of detection (POD) of 0.6 for a 2.0 volt repair limit. Since Duquesne Light Company has chosen to limit the voltage based plugging limit at 1.0 volt, EOC 11 SLB leakage is analyzed to be approximately 5 percent lower than the calculated SLB leakage with a 2.0 volt repair limit.

Therefore, implementation of the interim plugging criteria does not adversely affect steam generator tube integrity and implementation will be shown to result in acceptable dose consequences, therefore, the proposed amendment does not result in any 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?

Implementation of the proposed steam generator tube interim TSP plugging criteria does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism which could result in an accident outside of the region of the TSP elevations; no ODSCC that has been identified at the TSP has been detected outside the thickness of the TSPs. Neither a single or multiple tube rupture event would be expected in a steam generator in which the plugging criteria has been applied (during all plant conditions).

Specifically, Duquesne Light Company will implement a maximum leakage rate limit of 150 gpd per steam generator to help preclude the potential for excessive leakage during all plant conditions. The technical specification limits on primary-to-secondary leakage at operating conditions are to be a maximum of 450 gpd for all steam generators, or, a maximum of 150 gpd for any one steam generator. The RG 1.121 criterion for establishing operational leakage rate limits that require plant shutdown are based upon leak-before-break considerations to detect a free span crack before potential tube rupture during faulted plant conditions. The 150 gpd limit should provide for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. RG 1.121 acceptance criteria for establishing operating leakage limits are based on leak-before-break considerations such that plant shutdown is initiated if the leakage associated with the longest permissible crack is exceeded.

The single through wall crack lengths that result in tube burst at 1.43 times the steam line break pressure differential and SLB pressure differential alone are approximately 0.57 inch and 0.84 inch, respectively. A leak rate of 150 gpd will provide for detection of 0.41 inch long cracks at nominal leak rates and 0.62 inch long cracks at the lower 95 percent confidence level leak rates. Since tube burst is precluded during normal operation due to the proximity of the TSP to the tube and the potential exists for the crevice to become uncovered during SLB conditions, the leakage from the maximum permissible crack must preclude tube burst at SLB conditions. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for SLB conditions using the lower 95 percent leakage data. Additionally, this leak-before-break evaluation assumes that the entire crevice area is uncovered during blowdown. Partial uncovery will provide benefit to the burst capacity of the intersection. Analyses have shown that only a small percentage of the TSPs are deflected greater than the TSP thickness during a postulated SLB.

Steam generator tube integrity continues to be maintained through inservice inspection and primary-to-secondary leakage monitoring, therefore, the possibility of a new or different kind of accident from any accident previously developed is not created.

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

The use of the voltage based bobbin probe interim TSP elevation plugging criteria is demonstrated to maintain steam generator tube integrity commensurate with the requirements of RG 1.121. RG 1.121 describes a method acceptable to the NRC staff for meeting GDCs 14, 15, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the criteria, even under the worst case conditions, the occurrence of ODSCC at the TSP elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The EOC distribution of crack indications at the TSP elevations will be confirmed to result in acceptable primary-to-secondary leakage during all plant conditions and that radiological consequences are not adversely impacted.

In addressing the combined effects of loss of coolant accident (LOCA) and safe shutdown earthquake (SSE) on the steam generator component (as required by GDC 2), it has been determined that tube collapse may occur in the steam generators at some plants. This is the case as the TSP may become deformed as a result oflateral loads at the wedge supports at the periphery of the plate due to the combined effects of the LOCA rarefaction wave and SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse.

There are two issues associated with steam generator tube collapse. First, the collapse of steam generator tubing reduces the RCS flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase peak clad temperature (PCT). Second, there is a potential that partial through wall cracks in tubes could progress to through wall cracks during tube deformation or collapse.

Consequently, since the leak-before-break methodology is applicable to the BVPS reactor coolant loop piping, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design of the plant. The limiting LOCA event becomes either the accumulator line break or the pressurizer surge line break. LOCA loads for the primary pipe breaks were used to bound the conditions at BVPS for smaller breaks. The results of the analysis using the larger

break inputs show that the LOCA loads were found to be of insufficient magnitude to result in steam generator tube collapse or significant deformation. The LOCA and SSE tube collapse evaluation performed for another plant with Series 51 steam generators using bounding input conditions (large break loadings) is considered applicable to BVPS.

Addressing RG 1.83 considerations, implementation of the bobbin probe voltage based interim tube plugging criteria is supplemented by: enhanced eddy current inspection guidelines to provide consistency in voltage normalization, a 100 percent eddy current inspection sample size at the TSP elevations, and RPC inspection requirements for the larger indications left inservice to characterize the principal degradation as ODSCC.

As noted previously, implementation of the TSP elevation plugging criteria will decrease the number of tubes which must be repaired. The installation of steam generator tube plugs reduces the RCS flow margin. Thus, the implementation of the alternate plugging criteria will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to plant safety as defined in the Final Safety Analysis Report or any Bases of the 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, Pennsylvania 15001.

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

NRC Project Director: Walter R. Butler, Director

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

Date of amendment request: June 22, 1994

Description of amendment request: The proposed amendment revises technical specifications (TSs) related to the emergency feedwater system (EFW). The proposed changes extend the allowable outage time when one EFW train is inoperable from 36 hours to 72 hours and adapt other EFW sections from the "Restructured Standard Technical Specifications for B&W Plants" to the Arkansas Nuclear One, Unit 1 (ANO-1) format.

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

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

The Emergency Feedwater (EFW) system mitigates the consequences of any event with a loss of normal feedwater. This system is not the initiator of any previously analyzed accident, and therefore, changes to the specifications applicable to the EFW system present no significant increase in the probability of any previously evaluated accident.

The changes that revise the required Actions and Allowable Outage Times associated with the EFW system have been evaluated for their effect on the Core Damage Frequency (CDF) previously calculated in the ANO-1 Probabilistic Risk Assessment (PRA). The new ANO-1 CDF values, incorporating the proposed AOT extension, are 4.73E-05 (for the turbine-driven EFW pump) and 4.70E-05 (for the motor-driven pump). These values do not exceed the NRC Safety Goal of 1.0E-04 per reactor year, as stated in the Federal Register 50FR32138. The delta CDF associated with these changes (6.16E-07 for the turbine-driven EFW pump and 3.04E-07 for the motor-driven EFW pump) have been evaluated with respect to criteria contained in SECY-91-270, dated August 27, 1991, and NUMARC 91-04, dated January 1992, and fall within the category of events of low risk significance requiring no compensatory measures. This evaluation has shown the risk associated with the proposed changes to pose no undue risk to public health and safety, to be categorized as having low risk significance, and involve no significant increase in the consequences of an accident previously evaluated.

The changes revising the Limiting Conditions for Operation result in more restrictive controls on the operability of the motor-driven EFW pump. The previous specification required operability of both EFW pumps when the reactor was heated above 280°F. The proposed change requires the operability of the motor-driven EFW pump whenever the unit is above the cold shutdown condition and any steam generator is relied upon for heat removal. With this change, the motor-driven EFW pump is now required to be operable in a condition not previously specified, constituting an additional requirement not previously specified. This change does not involve a significant increase in the consequences of an accident previously evaluated.

The changes revising the Limiting Conditions for Operation also incorporate an Allowable Outage Time for the turbine-driven EFW pump steam supply valves which was not previously specified. The 7 day AOT is reasonable based on:

1. The redundant steam supply (from the opposite steam generator) to the turbine-driven EFW pump is operable,

2. The motor-driven EFW pump is operable, and

3. The probability of an event occurring that would require the inoperable steam supply valve to actuate is relatively low.

The changes to the surveillance specifications clarify the proper conditions required for the operability test of the turbine-driven EFW pump, and revise the requirement for the verification of proper EFW flow path valve alignment. The change clarifying the test conditions is required to ensure a sufficient steam supply to the turbine-driven EFW pump to perform the test. During plant startup, from an RCS temperature of 280°F to an RCS temperature of approximately 525°F (corresponding to a steam generator pressure of approximately 830 psig) the turbine-driven EFW pump is classified as available until operability is proven by successful completion of the surveillance requirement. The proposed changes state that the EFW pumps and their associated flow paths shall be operable when the RCS is above the cold shutdown condition with any steam generator relied upon for heat removal (motor-driven EFW pump) and when RCS temperature is greater than or equal to 280°F (turbine-driven EFW pump). This specification requires that the flow paths be properly aligned to maintain operability and is as restrictive as the current TS 4.8.1.c. The revised specification incorporates a new requirement to verify operator flexibility in determining the method of verification. Some methods that could be considered as fulfilling this requirement would include valve alignment checks, or a flow test verifying a level decrease in the 'Q' condensate storage tank with a corresponding level increase in both steam generators. These changes result in no significant increase in the consequences of an accident previously evaluated.

The other proposed changes included in this submittal, including the Bases changes, are considered to be administrative in nature and have no effect on the consequences of an accident previously evaluated. Relocation of the Emergency Feedwater Initiation and Control (EFIC) requirements from Section 3.4 to Section 3.5 places the requirements for this instrumentation system with the requirements for other instrumentation systems, resulting in greater consistency throughout the ANO-1 TS. Information in the Bases associated with the EFIC system has been corrected to reflect the actual plant condition and resolve a conflict with the ANO-1 Safety Analysis Report. The Bases changes add clarifying information to aid the operator in determining the applicability of the various EFW specifications.

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

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

The proposed changes introduce no new mode of plant operation. The EFW system is not an event initiator. It functions to mitigate the consequences of any event with a loss of normal feedwater.

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

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

The changes proposed to the Limiting Conditions for Operation associated with the EFW system are more conservative than the current specification, thus resulting in an increase in the margin of safety. The proposed changes to the actions required when both of the EFW trains are inoperable and the auxiliary feedwater pump is unavailable no longer require an immediate plant runback, that is currently required, which could introduce a plant transient, thus resulting in an increase in the margin of safety.

The changes revising the Limiting Conditions for Operations also incorporate and Allowable Outage Time for the turbine-driven EFW pump steam supply valves which was not previously specified. The 7 day AOT is reasonable based on:

1. The redundant steam supply (from the opposite steam generator) to the turbine-driven EFW pump is operable,

2. The motor-driven EFW pump is operable, and

3. The probability of an event occurring that would require the inoperable steam supply valve to actuate is relatively low.

The changes to the surveillance specifications clarify the proper conditions required for the operability test of the turbine-driven EFW pump, and revise the requirement for the verification of proper EFW flow path valve alignment. The change clarifying the test conditions is required to ensure a sufficient steam supply to the turbine-driven EFW pump to perform the test. During plant startup, from an RCS temperature of 280°F to an RCS temperature of approximately 525°F (corresponding to a steam generator pressure of approximately 830 psig) the turbine-driven EFW pump is classified as available until operability is proven by successful completion of the surveillance requirement. The proposed changes state that the EFW pumps and their associated flow paths shall be operable when the RCS is above the cold shutdown condition with any steam generator relied upon for heat removal (motor-driven EFW pump) and when RCS temperature is greater than or equal to 280°F (turbine-driven EFW pump). This specification requires that the flow paths be properly aligned to maintain operability and is as restrictive as the current TS 4.8.1.c. The revised specification incorporates a new requirement to verify proper alignment prior to relying upon any steam generator for heat removal. This allows the operator flexibility in determining the method of verification. Some methods that could be considered as fulfilling this requirement would include manual valve alignment checks, or a flow test verifying a level decrease in the 'Q' condensate storage tank with a corresponding level increase in both steam generators.

This change does involve an incremental reduction in the margin of safety since the extension of the EFW Allowable Outage Time from 36 hours to 72 hours does result in a slight increase in the Core Damage Frequency (CDF) as calculated in the ANO-1 Probabilistic Risk Assessment. The new ANO-1 CDF values, incorporating the proposed AOT extension, are 4.73E-05 (for the turbine-driven EFW pump) and 4.70E-05 (for the motor-driven EFW pump). These values do not exceed the NRC Safety Goal of 1.0E-04 per reactor year, as stated in the **Federal Register** 50FR32138. The CDF associated with these changes (6.16E-07 for the turbine-driven EFW pump and 3.04E-07 for the motor-driven EFW pump) have been evaluated with respect to criteria contained in SECY-91-270, dated August 27, 1991, and NUMARC 91-04, dated January 1992, and fall within the category of events of low risk significance requiring no compensatory measures. This reduction is not considered significant in that the increase in CDF has been evaluated as posing no undue risk to the public health and safety and is categorized as having low risk significance.

The other proposed changes included in this submittal, including the Bases changes, are considered to be administrative in nature. Relocation of the Emergency Feedwater Initiation and Control (EFIC) requirements from Section 3.4 to Section 3.5 places the requirements for this instrumentation system with the requirements for other instrumentation systems, resulting in greater consistency throughout the ANO-1 TS. Information in the Bases associated with the EFIC system has been corrected to reflect the actual plant condition and resolve a conflict with the ANO-1 Safety Analysis Report. The Bases changes add clarifying information to aid the operator in determining the applicability of the various EFW specifications.

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

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

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

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

NRC Project Director: William D. Beckner

Entergy Operations, Inc., Docket Nos. 50-313 and 50-368, ArkansasNuclear One, Unit Nos. 1 and 2 (ANO-1&2), Pope County, Arkansas

Date of amendment request: June 20, 1994

Description of amendment request: The proposed amendments revise the administrative and control sections of the technical specifications (TSs) for Arkansas Nuclear One, Units 1 and 2. The proposed changes relocate controls associated with the "Review and Audit" functions from the TSs to the Quality Assurance Program and relocate requirements for the audit of emergency and security plans and implementing procedures from the TSs to the respective emergency and security plans.

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

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

The proposed changes do not affect reactor operations or accident analyses, have no radiological consequences, and are considered to be purely administrative in nature. All requirements relocated from the TSs have been evaluated with respect to the four criteria of the NRC ≥Final Policy Statement On Technical Specifications Improvements" as presented in SECY-93-067, and found to meet none of the criteria for inclusion in the TS.

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

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

The proposed changes introduce no new mode of plant operation and do not affect the operability of safety-related equipment. All requirements relocated or deleted from the TSs have been evaluated with respect to the four criteria of the NRC "Final Policy Statement On Technical Specifications Improvements" as presented in SECY-93-067, and found to meet none of the criteria for inclusion in the TS.

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

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

Existing TS operability and surveillance requirements are not reduced by the proposed change, thus no margins of safety are reduced. All requirements relocated or deleted from the TSs have been evaluated with respect to the four criteria of the NRC "Final Policy Statement On Technical Specifications Improvements" as presented in SECY-93-067, and found to meet none of the criteria for inclusion in the TS.

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

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

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

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn,

1400 L Street, N.W., Washington, D.C. 20005-3502

NRC Project Director: William D. Beckner

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

Date of amendment request: February 9, 1993 as supplemented by letter dated July 22, 1994

Description of amendment request: The proposed amendment would revise Section 3.0 and 4.0 of the Technical Specifications (TSs) consistent with the provision and intent of Generic Letter (GL) 87-09 dated June 4, 1987.

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:

TS 3.0.4 prevents entry into an operational mode or other specified condition unless Limiting Conditions for Operations (LCOs) are met without reliance on Action Requirements. The intent of this TS is to ensure that a higher mode of operation is not entered when equipment is inoperable or when parameters exceed their specified limits.

The proposed change clarifies TS 3.0.4 such that LCOs with Action Statements that permit continued operation for an unlimited period of time are exempt from the restrictions of TS 3.0.4. This provision is modified to require an additional plant safety review prior to implementing additional exceptions to 3.0.4 other than those currently stated in the individual specifications. This proposed change is consistent with existing NRC regulatory requirements for LCOs.

The proposed change to TS 4.0.3 incorporates a 24-hour delay in implementing the Action Statements due to a missed surveillance requirement when the Action Statements provide a restoration time that is less than 24 hours. As reflected in GL 87-09, this change is justified in that it is overly conservative to assume that systems or components are immediately inoperable when a surveillance requirement has not been performed. The NRC concludes in Generic Letter 87-09 that a 24-hour time limit balances the risks associated with an allowance for completing the surveillance within this period against the risks associated with the potential for a plant upset and challenge to safety systems when the alternative is a shutdown to comply with Action Statements before the surveillance can be completed. The NRC further states that the potential for a plant upset and challenge to safety systems is increased if surveillances are performed during actions to initiate a shutdown to comply with Action Requirements.

TS 4.0.4 has been modified to note that its provisions shall not prevent passage through or to operational modes as required to comply with Action Requirements. This change is consistent with the intent of the existing TS and represents a clarification.

No previously analyzed accident scenario is changed by the proposed TS changes described above. Initiating conditions and assumptions remain as previously analyzed.

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

The proposed change to TS 3.0.4 is administrative in nature. Entry into an operational mode or other specified condition will be allowed for those specifications not currently stating an exception to 3.0.4 when 1) the applicable LCOs Action Requirement permits continued operation for an unlimited period of time and 2) the PORC [plant operations review committee] has reviewed and approved the exception.

The proposed change to TS 4.0.3 will allow continued operation for an additional 24-hours after discovery of a missed surveillance. As reflected in GL 87-09, missing a surveillance does not mean that a component or system is inoperable. In most cases, surveillances provide positive verification of operability.

The proposed change to TS 4.0.4 will alleviate conflict within the TS. The change is necessary to allow the plant to proceed through or to required operational modes to comply with Action Statements even if applicable Surveillance Requirements may not have been performed.

These changes do not affect the operation of the plant or the manner in which it is operated.

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

The proposed change to TS 3.0.4 is administrative in nature and will have no impact on any margin of safety.

The proposed change to TS 4.0.3 will allow up to 24 hours to perform a missed surveillance. In some cases this will eliminate the need for a plant shutdown. As reflected in GL 87-09, the overall effect is an increase in plant safety by avoiding unnecessary shutdowns and associated system transients due to missed surveillances.

The proposed change to TS 4.0.4 will eliminate an internal conflict within the TS and allow the plant to proceed through or to required operational modes to comply with Action Statements even if applicable Surveillance Requirements for that mode may not have been performed. The NRC staff has previously evaluated these change in Generic Letter 87-09 and determined that the TS modifications will result in improved TS.

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

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

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

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

NRC Project Director: William D. Beckner

Florida Power and Light Company, et al., Docket No. 50-335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of amendment request: July 28, 1994

Description of amendment request: The amendment will revise Technical Specifications (TS) 3/4.4.13 to incorporate Low Temperature Overpressure Protection (LTOP) requirements similar to those recommended by the NRC staff via Generic Letter 90-06. The proposed changes are in accordance with the resolution of Generic Issue 94 for St. Lucie Units 1 and 2.

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

Pursuant to 10 CFR 50.92, a determination may be made that a proposed license amendment involves no significant hazards consideration if 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. Each standard is discussed as follows:

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

The changes proposed for St. Lucie Unit 1 Technical Specifications (TS) 3/4.4.13 are similar to those recommended by the NRC staff via Generic Letter 90-06 for Low Temperature Overpressure Protection (LTOP) systems. On the basis of technical studies performed for Generic Issue 94, the staff concluded that LTOP system unavailability is a contributor to the risk associated with overpressure transients during the shutdown modes of plant operation. Revisions to the actions required and the time for completion of such actions, in the event that one or more Power Operated Relief Valves (PORV) become inoperable, provide more rigor than the existing specifications and are designed to increase LTOP system availability. The administrative restrictions do not change the results of existing analyses performed to evaluate postulated accidents but will improve the availability of systems designed to mitigate pressure transients that could

occur within the LTOP range. Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendment will not change the physical plant or the modes of operation defined in the facility license. The changes do not involve the addition of new equipment or the modification of existing equipment, nor do they alter the design of St Lucie plant systems. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendment provides additional administrative restrictions for the operation of LTOP equipment. The applicability of Limiting Conditions of Operation (LCO) involving the PORVs will be extended to include Operational MODE 6 when the head is on the reactor vessel, and the rigor of required actions and action compleiton times in the event that one or more PORVs become inoperable will be increased. Consequently, the risk of low temperature operations will be reduced and safety during the shutdown modes of operation will be enhanced. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Based on the discussion presented above and on the supporting Evaluation of Proposed TS Changes, FPL has concluded that this proposed license amendment involves no significant hazards consideration.

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

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

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

NRC Project Director: Victor M. McCree (Acting)

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

Date of amendment request: July 1, 1994

Description of amendment request: The proposed amendment to the Technical Specification (TS) would:1. Modify the facility by providing an auctioneered power supply for the engineered safety feature actuation system (ESFAS) sensor cabinets;2 Reinstate the 2-out-of-4 sump recirculation system (SRAS) logic;3. Change Table 3.3 of the (Safety Feature Actuation System Instrumentation) by adding Manual main steam isolation (MSI) (Trip Buttons); by removing note (f) which describes the SRAS logic as a modified 2-out-of-4 logic; and by replacing Action Statement 4 with an Action Statement that allows operation with a second inoperable channel, provided both channels are placed in the bypassed condition. 4. Add to the TS new limiting conditions for operation and new surveillance requirements together with BASES (TS 3.3.2.2 and 4.3.2.2.1 and 4.3.2.2.2) for the ESFAS sensor cabinet power supply drawers.

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 analyzed.

SRAS Logic Modification

Implementation of the auctioneered power supply for the sensor cabinets will permit the reinstatement of the original 2-out-of-4 (six possible combinations) logic for SRAS initiation. The current logic only has four possible combinations. Changing the minimum number of SRAS channels required to be operable from four to three does not significantly reduce the available actuation combinations. Operation with one channel inoperable will still provide a 2-out-of-3 logic (three possible trip combinations). With the current SRAS logic, operation with one channel in bypass does not meet the single failure criterion for proper SRAS operation. Amendment No. 168 prevents that condition.

Allowing continued operation with three operable channels is consistent with the original Millstone Unit No. 2 Technical Specifications (prior to Amendment No. 168).

Note (f), which describes the current logic, will no longer apply after the auctioneering circuit is installed. This note is for information only and has no associated action or surveillance requirements. Therefore, removal of note (f) cannot affect either the probability or consequences of a postulated accident.

In addition to the change in the minimum number of channels required to be operable, Action statement 4 will be revised to allow a limited period of two hours when a second channel may be placed in bypass for performance of surveillance testing. This is acceptable due to the installation of the auctioneering circuit and restoration of the full SRAS logic. Prior to the implementation of the short term modifications and Amendment No. 168, Action Statement 2 also applied to the SRAS. That Action Statement allows two hours of operation with two channels out of service. However, Action Statement 2 requires one of the two channels to be placed in the tripped position.

Postulating a LOCA (loss-of-coolant accident) and an additional failure, while in an action statement that specifies a maximum allowed outage time, is beyond the design basis of Millstone Unit No. 2. However, with one SRAS channel in bypass and one in the tripped position, an additional failure (such as the loss of a DC vital bus) following the onset of a LOCA could result in a false SRAS signal.

From an overall safety perspective, the potential consequences from a false SRAS at the onset of a LOCA are more severe than those from the failure to automatically generate an actuation signal. Proposed Action Statement 4 would require actuation of the remaining channel (following a LOCA and a loss of DC bus as a second failure) to initiate the SRAS. The existing operation procedures instruct the operator to ensure that the SRAS actuation occurs when the refueling water storage tank level decreases to a predetermined value. In the unlikely event that a LOCA occurred while a Action Statement 4 and no SRAS was generated at the appropriate time due to an additional failure which prevents one channel from tripping, the SRAS would be manually initiated by the operator.

The amount of time that Millstone Unit No. 2 would operate under Action Statement 4 (with two SRAS channels in bypass) is approximately 6 hours per month. This is based on the requirement to conduct monthly channel functional tests for the three operable channels. The probability of a LOCA occurring during these surveillance, while in Action Statement 4, with a subsequent failure of the remaining 2-out-of-2 SRAS logic, is very low.

Sensor Cabinet Auctioneering

The proposed new Technical Specification 3.3.2.2, which establishes the requirements for the ESFAS sensor cabinets power supply drawers, permits 48 hours to restore an inoperable sensor cabinet power supply drawer to operable status. A power supply drawer renders it inoperable, or if either its normal or backup power is not available.

Existing Technical Specification 3.8.2.1 contains an 8-hour action statement for restoring the power sources (VA-10, 20, 30 and 40) if they become inoperable. The proposed 48-hour action statement for the power supply drawers is appropriate since the sensor cabinet would remain functional if either normal or alternate power was not available. However, a LOCA and an additional failure while in the action statement could result in a false SRAS, since two channels would supplied from a single DC power supply.

Prior to Amendment No. 168, operation with an inoperable power supply drawer

could continue indefinitely, provided the provisions of Technical Specification 3/4.3.2 were followed. Operation with a power supply inoperable for an indefinite period of time places all the signals associated with that sensor cabinet in the tripped condition. This creates a 1-out-of-4 tripped condition for SRAS. In this condition, the single failure required to be postulated could result in a false SRAS actuation.

This 48-hour action statement is consistent with other action statements for ESFAS such as Action Statement 1 of Table 3.3-3. Also, this is consistent with the current wording of Action Statement 4 which allows 48 hours to restore an inoperable channel to operable while operating with the modified 2-out-of-4 logic.

MSI Trip Button Addition

The manual trip buttons provide a mechanism for the control room operator to initiate an MSI trip. The proposed Technical Specification change will require that a plant shutdown be initiated if either channel is out of service for more than 48 hours, and establishes a requirement for surveillance testing every refueling outage. Including the trip buttons in the Technical Specifications and establishing operation and surveillance requirements ensures their operability commensurate with their safety significance.

Based on the above, the changes to Technical Specification 3/4.3 do not increase the probability or consequence of an accident previously evaluated.

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

SRAS Logic Modification

Changing the number of channels required to be operable from four to three is acceptable since the original 2-out-of-4 logic will be restored. This change only affects the number and combinations of actuation channels necessary to initiate a SRAS. There is no change to the source or types of initiators, nor is there a change to the automatic response resulting from a SRAS.

Note (f), which described the modified logic, will no longer apply after the auctioneering circuit is installed. This note is for information only and has no associated action or surveillance requirements. Therefore, removal of note (f) cannot create a new or different kind of accident.

New Action Statement 4 restores the ability to operate for an indefinite period of time with one channel in bypass and for a limited period of time while two channels are out of service. The change from the original action statement to require that both channels be in bypass will prevent a false SRAS in the unlikely (and beyond design basis) event of a LOCA with an additional failure of a DC bus while in an LCO [limiting condition for operation].

Sensor Cabinet Auctioneering

The addition of a Technical Specification for the sensor cabinet power supply drawers does not create a potential for a new or different kind of accident. This new specification implements more restrictive operating requirements for the sensor cabinets. These are necessary to ensure that the sensor cabinets are energized from their primary power supply. The new specification does not affect the initiation of a SRAS signal nor the type of signal produced.

The auctioneering modification does bring two vital AC facilities together via isolation devices. This introduces a potential for a new type of failure mechanism. As described in Attachment 1, adequate isolation ensures that a failure on one side of an isolation transformer does not adversely degrade the other side.

MSI Trip Button Addition

The manual trip buttons provide a mechanism for the control room operation to initiate an MSI trip. The Technical Specification change will require that a plant shutdown be initiated if either manual trip channel is out of service for more than 48 hours, and establishes a requirement for surveillance testing every refueling outage. The trip buttons were installed during the 1992 outage. Establishing operability requirements and surveillance frequency cannot create a new or different kind of accident.

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

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

The net effect of the proposed modifications is to improve the reliability of the ESFAS and restore the design 2-out-of-4 logic for the SRAS. The proposed modifications improve the availability of the ESFAS, and do not affect the vital AC instrument panels.

The Technical Specification changes establish controls for the used of the SRAS with the restored logic configuration. The combination of the auctioneering of the power supplies, the restoration of the 2-out-of-4 logic, and the revised Technical Specifications restores the margin of safety and operational flexibility originally designed for the sensor cabinets.

Based on the above, 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 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: Gerald Garfield, Esquire, Day, Berry & Howard, City Place, Hartford, Connecticut 06103-3499.

NRC Project Director: John F. Stolz

PECO Energy Company, Docket No. 50-353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Date of amendment request: June 30, 1994

Description of amendment request: This amendment would remove certain remote shutdown system control valves and primary containment isolation valves from Technical Specifications Tables 3.3.7.4-1 and 3.6.3-1 respectively, as a result of eliminating the steam condensing mode of the Residual Heat Removal system.

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

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

These proposed changes will result in abandoning in place certain remote shutdown system control valves and removing from service and abandoning in place certain Primary Containment Isolation Valves (PCIVs) associated with the Residual Heat Removal (RHR) system steam condensing mode, and will remove the interface between the High Pressure Coolant Injection (HPCI) and RHR systems, therefore changing the primary containment pressure boundary.

The RHR system steam condensing mode is a non-safety related function of the RHR system; however, the pressure and structural integrity of the associated piping and valves are safety-related. These proposed changes will not affect any components required to perform the safety-related function of the RHR or HPCI systems.

The ability of the RHR or HPCI systems to respond to an accident will not be degraded. Only valves specifically dedicated for use for the RHR system steam condensing mode will be abandoned in-place, or removed from the plant. The valves' handswitches which are part of the remote shutdown panel (RSP) controls, will be physically removed from the RSP, since they will not perform any function (i.e., the associated valves will have the electrical power removed). The flanges and penetration caps that will become part of the primary containment boundary will be periodically tested for leakage as required by TS and 10CFR50, Appendix J. All piping and components that will remain operable will meet the original design requirements. The other modes of operation of the RHR system (e.g., Low Pressure Coolant Injection (LPCI), Shutdown (C)ooling (SDC)) will not be affected by these changes. Therefore, the proposed TS changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

No new failure modes of RHR or HPCI systems are created by the proposed TS changes. The proposed changes will have no impact on the existing High Energy Line Break (HELB) analysis for Limerick Generating Station (LGS). All valves or

piping removed and/or abandoned in place, are dedicated specifically for the RHR system steam condensing mode, and will not affect the operation of any components or piping required for other modes of operation of the RHR or HPCI systems. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The steam condensing mode is a non-safety related function of the RHR system and, therefore, is not addressed in the TS. This mode will be physically separated from the other modes of operation of RHR and HPCI systems, and consequently, will not preclude them from performing their safety-related functions. The remote shutdown system control valves to be abandoned in place are not being used presently, and the proposed changes will not impact the safety operation of LGS Unit 2. The primary containment penetration caps, safety-related pipe caps and the flanges replacing the removed PCIVs will be designed, fabricated and installed in accordance with the original design requirements, i.e., American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, 1971 Edition with Addenda through Winter 1971. The added penetration caps and flanges will be capable of maintaining the primary containment pressure boundary and isolation capabilities that were required of the PCIVs and will be tested for leakage periodically, as required by TS and 10 CFR 50, Appendix J. Additionally, all piping and components that will remain operable will meet original design requirements. Therefore, the proposed TS changes do not involve a reduction in a margin of safety.

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

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

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

NRC Project Director: Charles L. Miller

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

Date of amendment requests: July 19, 1994

Description of amendment request: This amendment will change the Technical Specification 3.1.5 for each unit for the standby liquid control system (SLCS) to remove the operability requirement for the SLCS while in Operational Condition 5 (refueling) with any control rod withdrawn, and to delete the 18-month system surveillance requirement (Surveillance Requirement 4.1.5.d.3).

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

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

The proposed Technical Specification change to delete the operability requirement for the SLC System in OPCON 5* (OPERATIONAL CONDITION 5 with any control rod withdrawn) does not affect the probability or consequences of an accident previously evaluated. Design basis accident mitigation scenarios for SSES in OPCON 5 do not depend on, or require, SLC operability; therefore, the proposed change to delete SLC operability in OPCON 5* does not affect the probability or consequences of an accident previously evaluated.

The proposed Technical Specification change to delete Surveillance Requirement 4.1.5.d.3, 18 month SLC heater operability check, does not affect the probability or consequences of an accident previously evaluated. Regarding the SLC heater function, the operability of the SLC system depends on maintaining the temperature of the sodium pentaborate solution above 70°F to prevent the boric acid from precipitating out of solution. SLC heater 'A' is used to maintain tank temperature between 85° F and 95°F, thus ensuring that the boric acid remains in solution. The operability of the heater 'A' is verified through the daily performance of Technical Specification Surveillance Requirement 4.1.5.a.1, which checks SLC solution temperature, and a control room alarm. Heater 'B' functions to raise SLC solution temperature prior to the mixing of SLC chemicals - the mixing of sodium pentaborate and water is an endothermic (heat consuming) reaction. The operability of heater 'B' is verified at the time when chemicals are added to the SLC tank, since a precondition for adding the chemicals is using heater 'B' to increase tank temperature to 100°F. Heater 'B' does not function to maintain tank temperature during normal operation. Therefore, the proposed change does not impact Susquehanna's ability to maintain SLC solution temperature and thus does not increase the probability or consequences of an accident previously evaluated.

2. This proposal does not create the possibility of a new or different kind of accident or [sic] from any accident previously evaluated.

The proposed Technical Specification change to delete the operability requirement for the SLC System in OPCON 5* does not create the possibility of a new or different kind of accident or [sic] from any accident previously evaluated. The purpose of the SLC System is to provide backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that none of the withdrawn control rods can be inserted. This bases is consistent with the required operability of the SLC System in OPCONs 1 & 2. The proposed change does not affect the ability of SLC to meet its design basis. No credit is taken for SLC in OPCON 5 to mitigate the effects of reactivity transients, and the SLC system is not designed to terminate an inadvertent criticality event during core alterations (OPCON 5) with vessel water level at least 22 feet above top of vessel flange. Therefore, no new or different accident scenarios are created by the proposed change.

The proposed Technical Specification change to delete Surveillance Requirement 4.1.5.d.3, 18 month SLC heater operability check, does not create the possibility of a new or different kind of accident or [sic] from any accident previously evaluated. The proposed change does not affect systems, structures, or components (SSCs) or the operation of these (SSCs). The heating and heater control subsystems of the SLC system will continue to function as they were designed. The proposed change does not alter the heating limits or the method for maintaining SLC solution temperature. Therefore, the proposed change does not create the possibility of a new or different kind of accident or [sic] from any accident previously evaluated.3. This change does not involve a significant reduction in a margin of safety.

The proposed Technical Specification change to delete the operability requirement for the SLC System in OPCON 5* does not involve a significant reduction in a margin of safety. The potential for a decrease in the margin of safety, under this proposed change, would be associated with periods during OPCON 5* when the SLC system was not operable. Allowing the SLC system to be inoperable during OPCON 5* with the vessel level at least 22 feet above top of vessel flange, represents no reduction in the margin of safety since the SLC System is not designed to terminate an inadvertent criticality event with a greater volume of water in the reactor. Having the SLC system inoperable in OPCON 5* with reactor water levels at normal operating volumes, does not significantly reduce the margin of safety because of the number of other design and operating features which act to prevent inadvertent criticality events. Adequate shutdown margin is maintained through design and administrative controls; including, Shutdown Margin Demonstration, Technical Specification 3.1.1, defueling and refueling procedures, and refueling interlocks. In addition, the Reactor Protection System monitors for recriticality and actuates the Control Rod Scram function if a significant reactivity addition is sensed.

The proposed Technical Specification change to delete Surveillance Requirement 4.1.5.d.3, 18 month SLC heater operability check, does not involve a significant reduction in a margin of safety. Adequate controls are in place, independent of the 18 month heater operability check, to ensure

that the temperature of the sodium pentaborate solution is maintained above 70° F. These controls include Surveillance Requirement 4.1.5.a.1, which checks SLC solution temperature daily, a control room alarm on low and high temperature, and the ambient temperature conditions in the SLC area which prevent rapid changes in SLC solution temperature. Operability of the 'B' heater is not needed to maintain SLC solution temperature, and the operability of this heater is verified at the time when chemicals are added to the SLC tank.

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: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701

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

NRC Project Director: Charles L. MillerPower Authority of the State of New York, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: June 13, 1994

Description of amendment request: The proposed amendment would modify the Facility Operating License by removing License Condition 2.E. This condition applies to the construction cleanup, restoration, and maintenance of transmission lines. It incorporated into the Facility Operating License the requirements of U.S. Department of Interior publication "Environmental Criteria for Electric Transmission Systems" - 1970. The proposed amendment was requested to eliminate duplication of regulatory authority by government agencies of the same activity.

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

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

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

The proposed change will remove a license condition unrelated to nuclear safety. License condition 2.E incorporated into the Operating License the requirements of U.S. Department of Interior publication "Environmental Criteria for Electric Transmission Systems" - 19^0. The goal of this standard is to "safeguard aesthetic and environmental values within the constraints imposed by the current state of high-voltage transmission technology." License condition 2.E addresses the preservation of the environment and natural resources. Removing this condition from the Facility Operating License has no bearing on plant safety or the health and safety of the public considering its non-nuclear safety nature. The transmission line right-of-ways maintained by the Authority are subject to regulation by other State and Federal agencies. Removal of this license condition will not affect operation of safety related structures, systems or components nor affect the quality assurance program at the FitzPatrick plant. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

License condition 2.E of the James A. FitzPatrick Plant Operating License applies to the construction cleanup, restoration, and maintenance of transmission lines. The Authority's transmission lines are managed under guidelines based on the "Generic Transmission Line Right-of-Way Management" plan requirements. The requirements imposed by the plan on the FitzPatrick transmission line right-of-ways exceed those of the U.S. Department of Interior publication referenced in license condition 2.E in both scope and details. Therefore, implementing the proposed change 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.

License condition 2.E of the James A. FitzPatrick Operating License applies to the construction cleanup, restoration, and maintenance of transmission lines. The requirements imposed by this license condition are unrelated to nuclear safety.

Continued operation of the plant without Condition 2.E does not involve a significant reduction in any margin of safety.

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

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

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

NRC Project Director: Michael L. Boyle

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

Date of amendment request: July 21, 1994

Description of amendment request: The proposed changes would modify paragraph 2.C.(3) of the Facility Operating License and relocate fire protection requirements from the Technical Specifications to an administrative procedure. These changes are based on the guidance contained in NRC Generic Letter 86-10, "Implementation of Fire Protection Requirements," and Generic Letter 88-12, "Removal of Fire Protection Requirements from Technical Specifications."

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

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

(1) This change does not involve a significant increase in the probability or consequences of an accident previously evaluated because no modifications, no changes to operating procedure requirements, no reduction in administrative controls and no reduction in equipment reliability are being made as a result of these changes. This proposed amendment relocates the fire protection LCOs [Limiting Conditions for Operation] and Surveillance Requirements from the Technical Specifications to an Administrative Procedure. No significant changes in content are being made to the Technical Specification requirements that are being relocated. Operating limitations will continue to be in effect, and required surveillances will continue to be performed in accordance with written procedures and instructions auditable by the NRC.

Although future proposed changes to the fire protection program elements previously located in the Technical Specifications will no longer be controlled by 10 CFR 50.36, proposed changes to the Fire Protection requirements will be controlled by the License Condition and plant procedures. Programmatic controls will continue to assure that fire protection program changes do not reduce the effectiveness of the program to achieve and maintain safe shutdown in the event of a fire.

(2) The possibility of an accident or malfunction of a different type than evaluated previously in the safety analysis report is not created because no reduction to the fire protection requirements, no modifications, no changes to operating procedure requirements, no reduction in administrative controls and no reduction in equipment reliability are being made as a

result of these changes. Programmatic controls will continue to assure that fire protection program changes do not reduce the effectiveness of the program to achieve and maintain safe shutdown in the event of a fire.

(3) This proposed amendment does not involve a reduction to the approved fire protection program or Fire Protection Technical Specification requirements because the Technical Specification fire protection requirements are being relocated, with no significant change in content, to an administrative procedure. Since there is no reduction in the requirements, no modifications, no changes to operating procedure requirements, no reduction in administrative controls and no reduction in equipment reliability are being made as a result of these changes, 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: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

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

NRC Project Director: Pao Tsin Kuo

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

Date of amendment request: July 25, 1994

Description of amendment request: The licensee has requested an amendment to the Technical Specifications (TS) to revise Table 3.6-1 (Non-Automatic Containment Isolation Valves Open Continuously or Intermittently for Plant Operation) and Table 4.4-1 (Containment Isolation Valves) to delete valves SI-1833A(B) and add valves SI-MOV-1835A(B). The valves being deleted no longer perform a containment isolation function as a result of a modification which removed the boron injection tank. The valves being added are needed for testing the safety injection 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:

Consistent with the criteria of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based on the following information:

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

Response:

The proposed license amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated. The change permits the removal of the two containment isolation valves on the Boron Injection Tank (BIT) bypass line. A previous amendment [Amendment No. 139, issued on October 15, 1993] to the Operating License removed the functional requirement for the BIT. Consequently, the function of the BIT bypass line to provide a Safety Injection (SI) pump test flow path has been rendered obsolete, permitting removal of the bypass line and associated valves. The bypass line will be cut and capped to assure containment integrity, therefore eliminating the need for containment isolation valves SI-1833A and SI-1833B. Opening the BIT outlet valve (SI-MOV-1835A or B) permits operability testing of the SI pumps, and is consistent with the current provision permitting opening of the BIT bypass valves. The changes do not impact the current operability and surveillance requirements for the Safety Injection System.

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

Response:

The proposed license amendment does not create the possibility of a new or different kind of accident from any previously evaluated. The change proposes to eliminate two containment isolation valves on the BIT bypass line whose function has been rendered obsolete by a previous amendment to the Operating License. The bypass line will be cut and capped to assure containment integrity, therefore eliminating the need for these containment isolation valves. Intermittent opening of the BIT outlet valve is consistent with the current provision permitting opening of the BIT bypass valves, thereby allowing operability testing of the SI pumps. The changes do not impact the operability or surveillance requirements for the Safety Injection System.

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

Response:

The proposed license amendment does not involve a significant reduction in a margin of safety for the following reasons. Currently, an orientation deficiency with the inboard BIT bypass isolation valve exposes its stem packing to the non-isolable side of the valve. The modification corrects this problem by removing both isolation valves and capping the pipes to assure integrity of the Containment and Safety Injection System. Additionally, removal of the isolation valves removes the potential for containment leakage resulting from valve degradation. Finally, removal of the BIT bypass line and its associated isolation valves does not inhibit the ability to test the SI pumps since a previous modification approved in an Amendment to the Operating License removed the functional requirement for the BIT. Consequently, the SI pumps may be flow tested with the BIT inservice, rendering obsolete the function of the BIT bypass line. Intermittent opening the BIT outlet valve is consistent with the current provision permitting opening of the BIT bypass valves, thereby allowing operability testing of the SI pumps.

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

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

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

NRC Project Director: Pao Tsin Kuo

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: June 29, 1994

Description of amendment request: These proposed amendments would revise the Technical Specifications to increase the minimum volume of oil contained in the Diesel Fuel Oil Storage Tanks (DFOSTs) at the Salem Generating Station (SGS). It would also revise the Updated Final Safety Analysis Report (UFSAR) description of the fuel oil storage system capability.

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

Emergency Diesel Generator (EDG) fuel oil is used to support mitigation of design basis events involving loss of the preferred (offsite) source of A.C. power. Fuel oil storage capacity has no effect on the probability of any accident previously evaluated.

Onsite fuel oil storage capability is designed to provide assurance of long term diesel operation to mitigate the consequences of a design basis accident. The proposed change would increase the minimum required volume in the Seismic Class I Diesel Fuel Oil Storage Tanks (DFOSTs), and would revise the Updated Final Safety Analysis Report (UFSAR), as part of an effort to reconstitute the basis for SGS fuel oil storage capacity. The DFOST inventory at the proposed minimum Technical Specification (TS) limit, combined with the emergency fill connection and Seismic Class III Fuel Oil

Storage Tank and transfer capability, would continue to provide a long term onsite fuel oil supply to the EDGs. Operations and Emergency Preparedness procedures would facilitate the transfer of fuel oil, and procurement from offsite sources as a contingency measure. Therefore, the ability to provide a long term supply of fuel oil to the EDG's is maintained, and the proposed change would not result in any significant increase in consequences of an accident previously evaluated.

(2) [This proposal does] not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change would increase the minimum DFOST level required by TS, and redefines the fuel oil storage and transfer systems' capability based on plant specific fuel oil consumption rate and EDG load profiles. These changes would not result in operation in any configuration prohibited by the present TS, and do not introduce the possibility of any new type of accident.

(3) [This change does] not involve a significant reduction in a margin of safety.

The EDG fuel oil storage and transfer capability would continue to support reliable, long term EDG operation, thereby maintaining an acceptable margin of safety relative to the ability of onsite A.C. power to support operation of equipment important to safety. The proposed changes do not involve a significant reduction in margin of safety.

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

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

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

NRC Project Director: Charles L. Miller

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

Date of amendment request: March 31, 1994 (TS 319)

Description of amendment request: The proposed amendment revises the setpoints for instrumentation used to isolate high energy line breaks in the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems. The proposed amendment also defines specific areas where steam line space temperatures are monitored.

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 to the HPCI and RCIC steam line space isolation setpoints do not affect any precursor for any design basis events or operational transients analyzed in the Browns Ferry Final Safety Analysis Report. Therefore, the probability of an accident previously evaluated is not increased.

The HPCI and RCIC steam line space high temperature isolations are provided to ensure automatic closure of each system's primary containment isolation valves for a HPCI or RCIC steam line break. The isolation occurs when a very small leak has occurred. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. As a result of the environmental qualification program, the environmental responses of the reactor building to high energy line breaks were analyzed. TVA used computer modeling techniques to predict the temperature response of various reactor building zones to high energy line breaks. The results indicate that the setpoints for the HPCI and RCIC temperatures switches should be lowered. The lower setpoints assure the timely initiation of a closure signal to the primary containment isolation valves. Therefore, assuring the maximum allowable temperatures are not exceeded.

The proposed change to the HPCI and RCIC steam line space isolation setpoints are in the conservative direction and provides the same or earlier detection and isolation of HPCI and RCIC steam line breaks.

The proposed trip level settings are high enough to ensure that spurious trips do not occur from normal or transient system operation and low enough to ensure that line breaks are detected and isolated before design conditions are exceeded. Therefore, the proposed changes will not significantly increase the consequences of an accident previously evaluated.

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

The proposed change to the HPCI/RCIC steam line space high temperature isolations does not involve any modification to plant equipment or changes in operating procedures. No new failure modes are introduced. There is no effect on the function or operation of any other plant system. No new system interactions have been introduced by the change. The results of a break in the HPCI or RCIC steam lines remain as before. The HPCI or RCIC steam line area temperature switches will still detect a break due to an increase in area temperature and provide an initiation signal to close the system primary containment isolation valves to prevent reactor coolant loss. The proposed change will conservatively serve to detect and mitigate HPCI and RCIC line breaks more expeditiously.

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

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

The proposed change will not reduce the margin of safety. The proposed change ensure that HPCI and RCIC steam line breaks are isolated at the same or lower steam line area temperatures. Computer modeling techniques were utilized to predict the temperature response in various areas through which the HPCI and RCIC steam lines pass. The revised setpoints are established above the maximum expected normal room temperatures to avoid spurious actions due to ambient conditions and below the analytical limits to ensure timely pipe break detection and isolation. Substantial margin exist between the maximum temperature expected in each area and the minimum actuation temperature determined for each temperature switch. With the substantial margin between maximum temperatures for the areas and the minimum actuation temperature of the switches, the maximum temperatures cannot result in actuation of the switches. The design and function of the affected components has not been changed. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Athens Public Library, South Street, Athens, Alabama 35611

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

NRC Project Director: Mr. Frederick J. Hebdon

Tennessee Valley Authority, Docket No. 50-260, Browns Ferry Nuclear Plant, Unit 2, Limestone County, Alabama

Date of amendment request: May 11, 1994 (TS 347T)

Description of amendment request: The proposed amendment extends the allowed outage time for the Browns Ferry Nuclear Plant (BFN) Unit 2 250 volt DC (direct current) control power supplies from 5 to 45 days. The amendment is a temporary revision to the BFN Unit 2 Technical Specifications (TS) to permit replacement of batteries and other hardware.

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 change involves temporarily (one-year period) extending the 5-day AOT [allowed outage time] for the 250-volt shutdown board control power supplies to 45 days. As such, this change does not increase the probability of any accident previously analyzed.

The 250-volt DC Power System is required to function to mitigate the consequences of design basis accidents. The loss of a single 250-volt DC shutdown board control power supply will result in a loss of control power for the 480-volt and the 4160-volt shutdown board that it serves. Loss of control power results in loss of only those engineered safeguards supplied by its respective shutdown boards. Redundant safe shutdown equipment exists to mitigate the consequences of design basis accidents. As discussed in Final Safety Analysis Report (FSAR) subsection 8.6.4.3, a single failure of a shutdown board control power supply is acceptable.

Loss of a single 250-volt plant DC power supply will not prevent Unit 2 safe shutdown. The 250-volt plant DC power supply system is designed so that any two out of the three power supplies carry the entire load needed for safe shutdown. As discussed in FSAR subsection 8.6.4.2 a single failure of a 250-volt plant DC power supply is acceptable.

At no time will control power be unavailable to the shutdown boards during the system upgrades. The proposed change will only increase the time allowed to operate the plant while a 250-volt DC shutdown board control power supply is out of service.

The proposed TS change allows an additional 40 days to perform system upgrades and results in a small increase in risk. This small increase in risk is associated with the probability and consequences of a 250-volt plant DC power supply malfunction while it is supplying shutdown board control power. The increase in risk associated with extending the AOT was analyzed in a Probabilistic Safety Assessment (PSA) and determined to be approximately 0.3 percent. This small increase in risk is determined to be insignificant and well within the uncertainty bounds of the PSA.

The proposed TS change does not change the function of any plant structure, system or component. The proposed change allows for improvements to the 250-volt DC shutdown board control power supply system. The improvements will increase the capability and reliability of the system. Qualified backup power will be utilized at all times during system modifications. Only one power supply will be out of service at a time during the modifications.

The small increase in risk is more than offset by the increased capability, capacity, and reliability of the new power supplies. Therefore, the power supply modifications will result in a net overall safety benefit.

[The licensee has also committed to implement compensatory measures while performing the power supply modifications. These measures provide additional confidence that potential accident consequences are not increased.]

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

Extending the 5-day AOT for the 250-volt shutdown board control power supplies to 45 days does not create the possibility of a new or different kind of accident, nor does it increase the probability that an accident will occur. The AOT extension does not involve plant modifications that could create the possibility of a new or different kind of accident from any of those discussed in the FSAR.

The 250-volt DC shutdown board control power supply modifications involve replacement of the existing components with more reliable, increased capacity equipment having the same functions as before.

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

The proposed TS change involves a risk increase of approximately 0.3 percent. TVA [the Tennessee Valley Authority, the licensee] considers this small increase to be insignificant. TVA also considers that the small increase in risk is offset by the benefits associated with replacing the control power supplies with new, upgraded equipment. Therefore, the proposed TS change does not involve a significant reduction in a margin of safety.

[The licensee has also committed to implement compensatory measures while performing the power supply modifications. These measures provide additional capability to mitigate an accident, minimizing any effect on safety margin.]

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

Local Public Document Room location: Athens Public Library, South Street, Athens, Alabama 35611

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

NRC Project Director: Mr. Frederick J. Hebdon

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

Date of amendment request: July 18, 1994

Description of amendment request: The proposed amendment would modify Point Beach Nuclear Plant Technical Specification (TS) 15.3.7, "Auxiliary Electrical Systems," by including an allowed outage time for one of the four connected station battery chargers and subsequent shutdown requirements. The basis for Section 15.3.7 would also be revised to support the above changes.

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

In accordance with the requirements of 10 CFR 50.91(a), Wisconsin Electric Power Company (Licensee) has evaluated the proposed changes against the standards of 10 CFR 50.92 and has determined that the operation of Point Beach Nuclear Plant, Units 1 and 2, in accordance with the proposed amendments, does not present a significant hazards consideration. A proposed facility operating license amendment does not present a significant hazards consideration if operation of the facility in accordance with the proposed amendment will not:

1. Create 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. Will not create a significant reduction in a margin of safety.

The proposed amendment allows operation for up to two hours with one out of the four connected station battery chargers out of service. The 2-hour outage time is based on Regulatory Guide 1.93 and reflects a reasonable time to assess plant status and either connect an operable battery charger to the affected DC bus or prepare to effect an orderly and safe shutdown of the operating unit(s). Since the batteries, chargers, and their associated vital instrument buses provide sufficient redundancy to assure the initiation of proper protective actions during degraded system conditions, operation of PBNP in accordance with these proposed amendments cannot create an increase in the probability or consequences of an accident previously evaluated, create a new or different kind of accident, or result in a significant reduction in a margin of safety. Therefore, the proposed changes do not present a significant hazards consideration.

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

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.

Georgia Power Company, Docket No. 50-366, Edwin I. Hatch Nuclear Plant, Unit 2, Appling County, Georgia

Date of amendment request: July 19, 1994

Description of amendment request: The proposed amendment would revise Technical Specification 3.3.6.6 to permit the traversing incore probe (TIP) system to be considered operable with less than four operable TIP units.Date of publication of individual notice in Federal Register: July 22, 1994 (59 FR 37516) Expiration date of individual notice: Comment Period expires August 8, 1994; Notice period expires August 22, 1994

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

Carolina Power & Light Company, Docket No. 50-261, H. B. Robinson SteamElectric Plant, Unit No. 2, Darlington County, South Carolina

Date of application for amendment: September 15, 1993

Brief description of amendment: The amendment revises the pressure-temperature limits from 15 to 24 effective full power years.

Date of issuance: July 29, 1994

Effective date: July 29, 1994

Amendment No. 149

Facility Operating License No. DPR-23. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: October 13, 1993 (58 FR 52980) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 29, 1994 No significant hazards consideration comments received: No

Local Public Document Room location: Hartsville Memorial Library, Home and Fifth Avenues, Hartsville, South Carolina 29550

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

Date of application for amendment: February 4, 1994

Brief description of amendment: The amendment revises the Action Statement of TS 3.6.5, Vacuum Relief System, to require in Mod... 1 4 with one vacuum relief system inoperable that the system be restored to the operable status within seventy-two hours or be in at least hot standby within the next six hours.

Date of issuance: July 27, 1994

Effective date: July 27, 1994

Amendment No. 49

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

Date of initial notice in **Federal Register**: March 30, 1994 (59 FR 14886) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 27, 1994.No significant hazards consideration comments received: No

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

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

Date of application for amendment: May 11, 1994

Brief description of amendment: The amendment revises TS 3/4.2.3 to establish limits on reactor power level as a function of total reactor coolant system (RCS) flow rate up to 5 percent below the current specified flow rate.

Date of issuance: July 27, 1994

Effective date: July 27, 1994

Amendment No. 50

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

Date of initial notice in **Federal Register**: May 25, 1994 (59 FR 27079) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 27, 1994.No significant hazards consideration comments received: No

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

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

Date of application for amendments: March 30, 1994, as supplemented by letters dated June 13, June 14, July 11, July 21 and July 28, 1994

Brief description of amendments: The amendment revises the Technical Specifications (TSs) by changing the Unit 1 heatup and cooldown pressure-temperature (P-T) curves (i.e., Figures 3.4-2a and 3.4-3a) to incorporate a newly determined reactor pressure vessel (RPV) reference nil-ductility temperature, RT_{NDT}. This new value of

RT_{NDT} was determined from the licensee's analysis of the first irradiation sample removed from Unit 1. The setpoint curve contained in Figure 3.4-4a for the Unit 1 Low Temperature Overpressure Protection System (LTOPS) is also revised to reflect the changes in the P-T curves and to provide a margin for uncertainties in measuring the reactor pressure. Additionally, the amendment updates the removal schedule of RPV surveillance capsules for both units in accordance with the American Society for Testing Materials (ASTM) Standard ASTM E185-82. Finally, the amendment incorporates an editorial change for Unit 2 in which some clarifying text was added in the Table of Contents to indicate the lifetime applicability of Figure 3.4-4b for Unit 2.

Date of issuance: July 29, 1994

Effective date: July 29, 1994

Amendment Nos.: 53 and 53

Facility Operating License Nos. NPF-72 and NPF-77. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 12, 1994 (59 FR 24747) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 29, 1994. No significant hazards consideration comments received: No

Local Public Document Room location: Wilmington Township Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Commonwealth Edison Company, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of application for amendments: September 10, 1993 as supplemented November 17, 1993

Brief description of amendments: The amendments revise the LaSalle County Station, Units 1 and 2 Updated Final Safety Analysis Report Section 11.5.2.1.4 to specify that operator action is required to trip the mechanical vacuum pump upon receipt of a main steam line high radiation alarm, rather than the action of an automatic trip, which is currently described in the UFSAR. NRC approval was required because the required operator action, an existing condition, is contrary to that described in the UFSAR and the NRC's Safety Evaluation Report related to the operation of LaSalle County station (NUREG-0519), and involved an unreviewed safety question.

Date of issuance: July 26, 1994

Effective date: July 26, 1994

Amendment Nos.: 101 and 85

Facility Operating License Nos. NPF-11 and NPF-18. The amendments revised the UFSAR.

Date of initial notice in **Federal Register:** December 1, 1993 (58 FR 63403) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 26, 1994. No significant hazards consideration comments received: No

Local Public Document Room location: Public Library of Illinois Valley Community College, Rural Route No. 1, Oglesby, Illinois 61348.

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

Date of application for amendments: June 16, 1994

Brief description of amendments: The amendments change specification 3/4.10.1 to recognize the exemption of a single valve on each unit from Type C testing until the next refueling outage on each unit.

Date of issuance: August 1, 1994

Effective date: August 1, 1994

Amendment Nos.: 155 and 143

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

Date of initial notice in **Federal Register:** June 30, 1994 (59 FR 33798) Public comments requested as to proposed no significant hazards consideration: yes. The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided an opportunity to request a hearing by August 1, 1994, 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 and final significant hazards consideration determination is contained in a Safety Evaluation dated August 1, 1994.

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

Consolidated Edison Company of New York, Docket No. 50-003 and Docket No. 50-247, Indian Point Nuclear Generating Unit Nos. 1 and 2, Westchester County, New York

Date of application for amendments: September 29, 1993

Brief description of amendments: The amendments revise the Technical Specifications (TSs) to change the submittal frequency of the Radioactive Effluent Release Report from semiannually to annually, and change the reporting date.

Date of issuance: July 21, 1994

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

Amendment Nos.: 44 and 172

Facility Operating License Nos. DPR-5 and DPR-26: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 24, 1993 (58 FR 62153) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 21, 1994. 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: September 29, 1993, as supplemented by letter dated April 1, 1994.

Brief description of amendment: The amendment revises the Technical Specifications to remove the cycle-specific parameter limits and to reference a Core Operating Limits Report containing these limits. These changes are in accordance with the guidance provided in Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications."

Date of issuance: July 26, 1994

Effective date: This license amendment is effective as of the date of issuance of the COLR by the licensee to be implemented no later than the return to operation following the 1995 refueling outage.

Amendment No.: 173

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

Date of initial notice in **Federal Register:** November 24, 1993 (58 FR 62154) The April 1, 1994, provided additional information that did not change the initial determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 26, 1994.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: January 28, 1994

Brief description of amendment: The amendment revises the TSs to change the containment isolation valve testing frequency and the acceptance criteria for the combined containment leakage rate to accommodate operation on a 24-month fuel cycle. These changes follow the guidance provided in Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle."

Date of issuance: July 29, 1994

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

Amendment No.: 174

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

Date of initial notice in **Federal Register:** April 13, 1994 (59 FR 17596) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 29, 1994.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: April 22, 1994, as supplemented July 6, 1994

Brief description of amendment: The amendment revised the reactor vessel pressure-temperature limits in the Technical Specifications. The change insures that the vessel fracture toughness requirements of Section V of 10 CFR Part 50, Appendix G, are satisfied through end of life.

Date of issuance: July 25, 1994

Effective date: July 25, 1994

Amendment No.: 113

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

Date of initial notice in **Federal Register:** May 12, 1994 (59 FR 24749). The July 6, 1994, letter provided clarifying information within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration findings. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 25, 1994.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 10, 1994

Brief description of amendment: The amendment revises the Fermi-2 Technical Specifications (TS) to remove Table 3.6.3-1, the list of primary containment isolation valves and Table 3.8.4.3-1, the list of safety systems' motor-operated valve thermal overload protection from the TS to administrative procedures in accordance with the guidance contained in Generic Letter 91-08.

Date of issuance: August 1, 1994

Effective date: August 1, 1994, with full implementation within 45 days.

Amendment No.: 102

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

Date of initial notice in **Federal Register:** June 8, 1994 (59 FR 29626) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 1, 1994.No significant hazards consideration comments received: No.

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

Duke Power Company, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: November 11, 1993, as supplemented on June 13, 1994

Brief description of amendments: The amendments revise the Technical Specification surveillance requirements for the emergency core cooling system subsystems.

Date of issuance: July 29, 1994

Effective date: July 29, 1994

Amendment Nos.: 145 and 127

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

Date of initial notice in **Federal Register:** April 13, 1994 (59 FR 17579) The June 13, 1994, letter provided clarifying and additional information that did not change the scope of the November 11, 1993, 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 July 29, 1994. No significant hazards consideration comments received: No.

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223

Duke Power Company, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: May 5, 1994, as supplemented June 13, 1994.

Brief description of amendments: The amendments revise the Technical Specifications to increase Main Steam and Pressurizer Code Safety Valve Setpoint Tolerances.

Date of issuance: August 2, 1994

Effective date: August 2, 1994

Amendment Nos.: 146 and 128

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

Date of initial notice in **Federal Register:** June 21, 1994 (59 FR 32029) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 2, 1994.No significant hazards consideration comments received: No.

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223

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

Date of amendment request: December 14, 1993

Brief description of amendment: The amendment revised the Technical Specifications to revise the azimuthal power tilt limit from less than or equal to 0.10 (10%) to less than or equal to 0.03 (3%) and revises the action statement for control element assembly misalignment to allow 24 hours to restore the tilt to less than 3%.

Date of issuance: August 3, 1994

Effective date: August 3, 1994

Amendment No.: 97

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

Date of initial notice in **Federal Register:** January 19, 1994 (59 FR 2866) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 3, 1994.No significant hazards consideration comments received: No.

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

GPU Nuclear Corporation, et al., Docket No. 50-219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of application for amendment: April 6, 1994, as supplemented June 21, 1994

Brief description of amendment: The amendment eliminates the scram and main steam line isolation valve (MSIV) closure requirements associated with the main steam line radiation monitors (MSLRM). The amendment also eliminates the following related automatic isolation functions that are associated with the MSLRM scram and MSIV isolation: a) Main Steam Line Condenser Drain Valves, b) Emergency Condenser Drain Valves, c) Reactor Recirculation Loop Sample Valve, d) Instrumental Air Valves, and e) Condenser Pump Isolation.

Date of issuance: July 29, 1994

Effective date: As of the date of issuance to be implemented at the restart from refueling outage 15R.

Amendment No.: 169

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

Date of initial notice in **Federal Register:** April 28, 1994 (59 FR 22008). The June 21, 1994, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated July 29, 1994.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 15, 1993.

Brief description of amendment: The amendment revises the plant Technical Specifications (TSs) on the Reactor Coolant Inventory Trending System (RCITS). The change is consistent with NUREG-1430 entitled "Standard Technical Specifications for Babcock and Wilcox Plants." The RCITS information will be available to the operator to enhance the operator's ability to understand and manage transients and events when needed.

Date of issuance: August 1, 1994

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

Amendment No.: 191

Facility Operating License No. DPR-50. Amendment revised the Technical Specifications. I11*Date of initial notice in* **Federal Register:** June 8, 1994 (59 FR 29626) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 1, 1994. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: February 10, 1994

Brief description of amendment: The amendment revises the TMI-1 Technical Specifications (TS) to revise specification 3.7.2.c, "Unit Electric Power System," to provide an option to testing an emergency diesel generator (EDG) when the redundant EDG is inoperable.

Date of issuance: July 25, 1994

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

Amendment No.: 188

Facility Operating License No. DPR-50. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 22, 1994 (59 FR 32230) The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated July 25, 1994.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 2, 1994

Brief description of amendment: The amendment revises the plant Technical Specifications to modify Operational Safety Instrumentation requirements to specify completion time which allows for performance of maintenance or surveillance within a reasonable time and to be consistent with the allowable outage time for other safety-related equipment when only one train is affected.

Date of issuance: July 25, 1994

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

Amendment No.: 189

Facility Operating License No. DPR-50. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 13, 1994 (59 FR 17600). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 25, 1994.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 11, 1994

Brief description of amendment: The amendment revises the plant Technical Specifications to specify an allowable outage time for the Emergency Feedwater Pumps during surveillance activities. It also changes the requirement to test redundant components for operability to a requirement to ensure operability based on verification of completion of appropriate surveillance activities.

Date of issuance: July 25, 1994

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

Amendment No.: 190

Facility Operating License No. DPR-50. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 13, 1994 (59 FR 17601) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 25, 1994.No significant hazards consideration comments received: No

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

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

Date of amendment request: January 14, 1994

Brief description of amendment: The amendment revised TS Sections 3/4.3, "Instrumentation," 3/4.4.2, "Safety/Relief Valves," and associated Bases to increase the surveillance test intervals and allowable out-of-service times for various instruments.

Date of issuance: August 2, 1994

Effective date: August 2, 1994

Amendment No.: 74

Facility Operating License No. NPF-47. The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 26, 1994 (59 FR 21787) The additional information contained in the supplemental letter dated July 15, 1994, was clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 2, 1994.No significant hazards consideration comments received. No.

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

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: March 16, 1994

Brief description of amendments: The amendments modified Figure 3.4-4, "Nominal Maximum Allowable PORV Setpoint for the Cold Overpressure System," for the cold overpressure mitigation system with a revised setpoint curve.

Date of issuance: August 3, 1994

Effective date: August 3, 1994, to be implemented within 31 days of issuance

Amendment Nos.: Unit 1 - Amendment No. 63; Unit 2 - Amendment No. 52

Facility Operating License Nos. NPF-76 and NPF-80. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 13, 1994 (59 FR 17601) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 3, 1994.No significant hazards consideration comments received: No

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, Texas 77488

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

Date of application for amendment: April 29, 1994

Brief description of amendment: The amendment changes the requirement for reactor operators in Table 6.2-1 from 2 to 3 for the RUN, STARTUP/HOT STANDBY and HOT SHUTDOWN conditions. In addition, two typographical corrections were made to page 6-4.

Date of issuance: August 2, 1994

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

Amendment No.: 75

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

Date of initial notice in **Federal Register**: June 22, 1994 (59 FR 32231) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 2, 1994. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 8, 1993 (Reference HBL-93-058)

Brief description of amendment: This amendment modified the Technical Specifications (TS) incorporated in *Facility Operating License No.* DPR-7 as Appendix A by incorporating a title change into Section VII, Administrative Controls. This change reflects a plant organizational name change.

Date of issuance: July 26, 1994

Effective date: This license amendment is effective as of the date of its issuance and must be fully implemented no later than 30 days from the date of issuance.

Amendment No.: 27

Facility Operating License No. DPR-7: The amendment revised the TS.

Date of initial notice in **Federal Register**: January 5, 1994 (59 FR 624) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 26, 1994No significant hazards consideration comments received: No.

Local Public Document Room location: Humboldt County Library, 636 F Street, Eureka, California 95501

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

Date of application for amendments: October 27, 1993, as supplemented by letters dated April 29, 1994, and June 27, 1994

Brief description of amendments: These amendments revise the Unit 2 and Unit 3 Technical Specifications to allow one of the required on-shift senior reactor operators (SRO) to be combined with the required shift technical advisor (STA) position (i.e., dual-role SRO/STA position) as long as a minimum of three qualified individuals fill the SRO and STA positions.

Date of issuance: August 2, 1994

Effective date: August 2, 1994

Amendments Nos.: 191 and 196

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

Date of initial notice in **Federal Register**: December 8, 1993 (58 FR 64613)The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 2, 1994.No significant hazards consideration comments received: No

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

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

Date of application for amendment: May 10, 1994

Brief description of amendment: The Technical Specifications amendment revised Section 3.1.C.3 and Table 4.1-1 of Appendix A of the Operating License. These changes require that the reactor coolant average temperature (T_{avg}) be no lower than 540°F during critical operation. Critical operation at T_{avg} less than 540°F requires operator response to restore T_{avg} to greater than or equal to 540°F within 15 minutes or be in hot shutdown within the following 15 minutes. Additionally, the change in Table 4.1-1 entitled, "Minimum Frequencies for Checks, Calibrations and Tests," adds the requirement for T_{avg} instrument check frequency to be reduced to 30 minutes when the T_{avg} banks are above zero steps.

Furthermore, the revision to the Bases indicates that the minimum temperature for criticality provides assurance that the reactor is operated within the bounds of the safety analyses. Also included is an administrative change to correct some typographical errors on page 3.1-25 of the Technical Specifications.

Date of issuance: July 25, 1994

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

Amendment No.: 149

Facility Operating License No. DPR-64: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 8, 1994 (59 FR 29630) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 25, 1994. No significant hazards consideration comments received: No

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

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

Date of application for amendments: May 3, 1994

Brief description of amendments: The amendments revised the Technical Specification (TS) for Combustible Gas Control (3/4.6.4.1) by changing the surveillance frequency for performing the channel functional test to once-per-quarter and the channel calibration to once-per-refueling. Also, the TS for the Auxiliary Feedwater System (3/4.7.1.2) were changed to reduce the surveillance frequency for performing pump operability tests to once-per-quarter on a staggered test basis. These changes are consistent with the provisions of Generic Letter 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements For Testing During Power Operations."

Date of issuance: July 27, 1994

Effective date: July 27, 1994

Amendment Nos. 153 and 134

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

Date of initial notice in **Federal Register:** June 8, 1994 (59 FR 29634) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 27, 1994.No significant hazards consideration comments received: No

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

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

Date of application for amendment: March 11, 1994

Brief description of amendment: The amendment changes the Technical Specifications to delete TS Surveillance Requirement 4.8.4.1.a.3 that requires periodic retest of containment penetration overcurrent protection fuses and to remove references to containment penetration fuse testing from the TS Bases.

Date of issuance: July 29, 1994

Effective date: July 29, 1994

Amendment No.: 115

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

Date of initial notice in **Federal Register:** May 12, 1994 (59 FR 24752) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 29, 1994.No significant hazards consideration comments received: No

Local Public Document Room location: Fairfield County Library, Garden and Washington Streets, Winnsboro, South Carolina 29180.

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

Date of application for amendment: June 17, 1994

Brief description of amendment: The amendment changes the Technical Specifications to revise the nuclear enthalpy rise hot channel factor (F delta H) from equal to or less than 1.65 [1 plus 0.3(1-P)] to equal to or less than 1.70 [1 plus 0.3(1-P)] where P is a fraction of rated power. The amendment also revises the action statement to reflect guidance contained in the improved standard technical specifications.

Date of issuance: July 22, 1994

Effective date: July 22, 1994

Amendment No.: 109

Facility Operating License No. NPF-2 Amendment revises the Technical Specifications

Date of initial notice in **Federal Register:** June 22, 1994 (59 FR 32249) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 22, 1994.No significant hazards consideration comments received: No

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

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

Date of application for amendments: August 25, 1992 (TS 321)

Brief description of amendments: The amendments delete reference to recirculation equalizer valves from the technical specifications. These components have been removed from Browns Ferry Unit 3, and are not used in Browns Ferry Units 1 and 2.

Date of issuance: August 4, 1994

Effective date: August 4, 1994

Amendment Nos.: 211, 226 and 184

Facility Operating License Nos. DPR-33, DPR-52 and DPR-68: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** October 28, 1992 (57 FR 48829) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 4, 1994.No significant hazards consideration comments received: None

Local Public Document Room location: Athens Public Library, South Street, Athens, Alabama 35611

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: June 17, 1993 (TS 93-08)

Brief description of amendments: The amendments revise the allowable values for the intermediate and source range neutron flux reactor trip setpoints.

Date of issuance: July 26, 1994

Effective date: July 26, 1994

Amendment Nos.: 185 and 177

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

Date of initial notice in **Federal Register:** August 4, 1993 (58 FR 41514) The Commission's related evaluation of the amendments are contained in a Safety Evaluation dated July 26, 1994.No significant hazards consideration comments received: None

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

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

Date of application for amendment: February 17, 1993

Brief description of amendment: This amendment increases the TS trip setpoint and its associated allowable value for containment high-radiation specified in TS Table 3.3-4 from "<2 x Background at RATED THERMAL POWER" to "<4 x Background at RATED THERMAL POWER."

Date of issuance: July 27, 1994

Effective date: As of the date of issuance to be implemented within 90 days.

Amendment No. 190

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

Date of initial notice in **Federal Register**: June 23, 1993 (58 FR 34095) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 27, 1994. No significant hazards consideration comments received: No

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

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

Date of amendment request: February 14, 1994, as supplemented by letter dated April 29, 1994.

Brief description of amendments: The amendments increase the boron concentration limits for the Unit 2 refueling water storage tank and emergency core cooling system, and delete a footnote concerning refueling canal boron concentration during initial fuel load for both units.

Date of issuance: August 2, 1994

Effective date: August 2, 1994, to be implemented prior to startup for Cycle 2 for Comanche Peak Steam Electric Station, Unit 2.

Amendment Nos.: Unit 1 - Amendment No. 26; Unit 2 - Amendment No. 12

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

Date of initial notice in **Federal Register**: April 28, 1994 (59 FR 22015) The information contained in the April 29, 1994, letter was editorial in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 2, 1994.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: February 17, 1994, supplemented by letter dated May 18, 1994

Brief description of amendment: The amendment revises the Technical Specification 3/4.5.1 and associated Bases Section 3/4.5.1. A new Action Statement a. provides a 72-hour allowed outage time (AOT) for one accummulator inoperable due to boron concentration. The Action Statement b. AOT was changed to 24 hours. Surveillance Requirements 4.5.1.1.a.1 and 4.5.1.1.b were revised and 4.5.1.2 was deleted from the TS.

Date of issuance: August 5, 1994

Effective date: August 5, 1994 to be implemented within 30 days

Amendment No.: 91

Facility Operating License No. NPF-30. Amendment revised the Technical Specifications 3/4.5.1 and associated Bases Section 3/4.5.1.

Date of initial notice in **Federal Register**: March 30, 1994 (59 FR 14898) The additional information contained in the May 18, 1994, letter provided additional supplemental 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 August 5, 1994.No significant hazards consideration comments received: No.

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

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

Date of application for amendments: April 19, 1994

Brief description of amendments: The amendments revise the NA-1&2 Technical Specifications surveillance frequency requirements for control rod motion testing from once per 31 days to once per 92 days.

Date of issuance: July 28, 1994

Effective date: July 28, 1994

Amendment Nos.: 185 and 166

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

Date of initial notice in **Federal Register**: May 25, 1994 (59 FR 27070) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 28, 1994.No significant hazards consideration comments received: No.

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

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

Date of application for amendments: April 19, 1994

Brief description of amendments: These amendments modify the surveillance frequency of the control rod motion testing from monthly to quarterly.

Date of issuance: August 2, 1994

Effective date: August 2, 1994

Amendment Nos. 192 and 192

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

Date of initial notice in **Federal Register**: May 25, 1994 (59 FR 27070) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 2, 1994No significant hazards consideration comments received: No

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

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

Date of application for amendment: April 1, 1993

Brief description of amendment: The amendment modifies the Technical Specifications to add inservice inspection requirements for reactor coolant system piping in accordance with Generic Letter 88-01, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping." In addition, the amendment corrects an administrative error in a TS that references a table listing high/low pressure interface valve leakage pressure monitors.

Date of issuance: July 28, 1994

Effective date: 30 days after the date of issuance.

Amendment No.: 130

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

Date of initial notice in **Federal Register**: May 12, 1993 (58 FR 28065) The Commission's related evaluation of the amendment is contained in a Safety

Evaluation dated July 28, 1994. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: February 17, 1994, supplemented by letter dated May 13, 1994.

Brief description of amendment: The amendment changes the Technical Specifications (TS) 10-year hydrostatic testing requirements. The changes (1) add a special test exception for inservice leak testing and hydrostatic testing, (2) add a new minimum reactor vessel metal pressure-temperature curve for less than or equal to eight effective full power years, and (3) delete Table B 3/4.4.6-1, "Reactor Vessel Toughness," from the TS bases.

Date of issuance: May 27, 1994

Effective date: May 27, 1994

Amendment No.: 122

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

Date of initial notice in **Federal Register:** March 30, 1994 (59 FR 14902) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 27, 1994. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: December 1, 1993

Brief description of amendment: The amendment revises the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS) by incorporating technical and administrative changes to TS 3.10, Control Rod and Power Distribution Limits. The changes eliminate specifications for fuel designs no longer used at Kewaunee, specify required actions to be taken upon exceeding control bank insertion limits, and revise the limits for Departure from Nucleate Boiling (DNB) related parameters to assure operation within the assumptions of the Updated Safety Analysis Report (USAR) analyses.

Date of issuance: August 3, 1994

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

Amendment No.: 110

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

Date of initial notice in **Federal Register:** February 2, 1994 (59 FR 4949) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 3, 1994. No significant hazards consideration comments received: No.

Local Public Document Room location: University of Wisconsin Library Learning Center, 2420 Nicolet Drive, Green Bay, Wisconsin 54301.

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

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

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

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

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

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

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

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

For further details with respect to the action see (1) the application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555, and at the local public document room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By September 16, 1994, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555 and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

Date of application for amendments: August 2, 1994

Brief description of amendments: The amendments revise the Technical Specifications by adding a footnote that recognizes that through the end of cycle 6, the Unit 1, loop B wide range hot leg indication at the remote shutdown panel is inoperable.

Date of issuance: August 5, 1994

Effective date: August 5, 1994

Amendment Nos.: 63 and 63

Facility Operating License Nos. NPF-37 and NPF-66. The amendments revised the Technical Specifications.Public comments requested as to proposed no significant hazards consideration: No. The Commission's related evaluation of the amendments, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated August 5, 1994.

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

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

NRC Project Director: Robert A. Capra

Dated at Rockville, Maryland, this 10th day of August 1994.

For The Nuclear Regulatory Commission

Steven A. Varga,

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

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