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

OFFICE OF THE
COMMISSIONER

April 1, 1985

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MEMORANDUM FOR: Nunzio J. Palladino, Chairman

FROM: Lando W. Zech, Jr. Lando W. Zech Jr.

SUBJECT: REGULATIONS IMPLEMENTING SHOLLY AMENDMENT

There is considerable discussion on our implementation of the Sholly Amendment in the recent Investigative Staff Report to the House Appropriations Committee. In looking into the background, I learned that NRC published two interim final rules in May, 1983. It is my understanding that we solicited comments on these interim rules and the staff submitted a proposed final rule sometime thereafter.

Now that almost two years have gone by since the publication of the interim rules, I would appreciate a status report on this project. I want to be assured that our procedures implementing the Sholly Amendment are not using resources unnecessarily because of an overly-restrictive interpretation of the requirements of that amendment.

cc: Commissioner Roberts
Commissioner Asselstine
Commissioner Bernthal
SECY
EDO
OPE
OGC
ELD

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federal register

Wednesday
March 27, 1985

Part III

Nuclear Regulatory Commission

Applications and Amendments to Operating Licenses Involving No Significant Hazards Considerations; Monthly Notice

NUCLEAR REGULATORY COMMISSION

Applications and Amendments to Operating Licenses Involving No Significant Hazards Considerations; Monthly Notice

I. Background

Pursuant to Public Law (Pub. L.) 97-415, the Nuclear Regulatory Commission (the Commission) is publishing its regular monthly 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 monthly notice includes all amendments issued, or proposed to be issued, since the date of publication of the last monthly notice which was published on February 27, 1985 (50 FR 7979) through March 18, 1985.

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION AND OPPORTUNITY FOR 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 amendments 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. The Commission will not normally make a final determination unless it receives a request for a hearing.

Comments should be addressed to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch.

By April 26, 1985 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 petition for leave to intervene. Requests for a hearing and petitions 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. 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 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 fifteen (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 fifteen (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, and the bases for each contention set forth with reasonable specificity. Contentions shall be limited to matters within the scope of the amendment under consideration. 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 amendement 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 involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

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 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.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commisison, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C., by the above date. Where petitions are filed during the last ten (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 (800) 325-6000 (in Missouri (800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to *(Branch Chief)*: 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 Executive Legal Director, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board designated to rule on the petition and/or request, that the petitioner has made a substantial showing of good cause for the granting of a late petition and/or request. That determination will be based upon a balancing of the 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, 1717 H Street, NW., Washington, D.C., and at the local public document room for the particular facility involved.

Arkansas Power & Light Company, Docket No. 50-368, Arkansas Nuclear One, Unit 2, Pope County, Arkansas

Date of amendment request: December 21, 1984.

Description of amendment request: The proposed amendment would revise the steam generator low water level trip setpoints specified in Table 2.2-1 and Table 3.3-4 of the Technical Specifications (TS). Specifically, the reactor protective instrument trip setpoint and the Engineered Safety Feature Actuation System (ESFAS) trip value for the steam generator low water level would be reduced from 46.7% to 23%. Similarly, the allowable values in these tables would be reduced by the same magnitude from 45.811% to 22.111%. Reducing these setpoints is expected to reduce the probability of unnecessary reactor trips during certain planned operating maneuvers, such as manual control of steam generator water levels at low power.

The purpose of the steam generator low water level reactor trip is to provide protection against a loss of normal feedwater flow incident. The reactor trip setpoint should provide allowance that there will be sufficient water inventory in the steam generators at the time of the trip to provide sufficient margin before emergency feedwater is required. Automatic actuation of the Emergency Feedwater System (EFWS) is initiated when several parameters, including the steam generator water level, reach the ESFAS trip values.

Basis for proposed no significant hazards consideration determination: The loss of normal feedwater flow is analyzed in Chapter 15 of the ANO-2 Updated Final Safety Analysis Report (UFSAR). There, the setpoint for the steam generator low water level used in the accident analyses is 5%. The applicable ESFAS trip value is also 5% in the accident analyses. The results of the loss of normal feedwater flow analysis show that the plant protection system consisting of the Reactor Protective System (RPS) and the ESFAS will assure that the fuel design limit is not exceeded and that the steam generator heat removal capability is maintained in the event of a loss of normal feedwater flow. The analyses setpoint of 5%, when corrected for equipment errors and measurement uncertainties, results in a proposed setpoint of 23%.

In the December 21, 1984 application, the licensee states that the present trip setpoint was selected during the initial licensing review of ANO-2 in order to resolve questions concerning asymmetric steam generator events. After obtaining an operating license, the licensee modified its Core Protection Calculators (CPC) software to include cold leg temperature difference bias algorithm to provide a reactor trip in the event of an asymmetric steam generator transient. This modification was reviewed and approved by the NRC staff in its Safety Evaluation dated June 19, 1981.

In addition to our preliminary review of the loss of normal feedwater flow event and our review of the asymmetric steam generator transient, we performed a preliminary review of all other events in Chapter 15 of the ANO-2 UFSAR. No adverse effects resulting from the proposed changes have been identified in our review. Therefore, the change is clearly within all acceptable criteria with the Reactor Coolant System and its associated auxiliaries as contained in Section 15.2.7 of the Standard Review Plan (SRP), "LOSS OF NORMAL FEEDWATER FLOW" which is the applicable section of the SRP for the systems involved.

Therefore, the proposed changes match an example of "no significant hazard" in the guidance provided by the Commission (48 FR 14870), namely, a change which "may reduce in some way a safety margin but where the results of the change are clearly within all acceptable criteria with respect to the system as specified in the Standard Review Plan." Thus, the staff proposes to determine that the application involves no significant hazards consideration.

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

Attorney for licensee: Nicholas S. Reynolds, Esq., Bishop, Liberman, Cook, Purcell and Reynolds, 1200 Seventeenth Street, NW., Washington, D.C. 20036.

NRC Branch Chief: James R. Miller.

Arkansas Power & Light Company, Docket No. 50-368, Arkansas Nuclear One, Unit 2, Pope County, Arkansas

Date of amendment request: January 28, 1985.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to remove the rod bow penalty factor surveillance requirement. Specifically, TS 4.2.4.4 which requires that certain DNBR penalty factors shall be verified to be included in the COLSS and CPC DNBR calculations periodically would be deleted.

The DNBR (Departure from Nucleate Boiling Ratio) is a unitless value calculated from reactor core thermal-hydraulic conditions on a real-time basis from an NRC approved empirical correlation. It is a measure of thermal margin. Maintaining core conditions such that DNBR is above a prescribed value ensures that the fuel cladding will not overheat during normal and abnormal plant operation. The CPC (Core Protection Calculators), which are an integral part of the reactor protection system (RPS) at ANO-2, monitor certain NSSS variables and initiate a reactor trip if fuel design limits are approached as a result of an abnormal event. The COLSS (Core Operating Limit Supervisory System) is a monitoring system which continuously calculates and advises operators of margins to core operating limits on fuel design and the licensed power level. The COLSS provides an alarm if any one of the core operating limits is exceeded.

In Supplement No. 1 to the Safety Evaluation Report (NUREG-0308) of June 1978 for the issuance of the ANO-2 operating license, the NRC staff required that certain conservative DNBR penalty factors due to rod bowing as functions of fuel burnups be used in DNBR calculations. The above requirement was imposed since ANO-2 was the lead plant with Combustion Engineering (CE) 16x16 fuel design; therefore, there was no irradiated fuel data germane to the CE 16x16 fuel design at the time.

Since the issuance of the ANO-2 operating license, CE has accumulated and studied irradiated fuel data specific to ANO-2. A CE report, CEN-289(A), provides the results of the CE study. The report, which was submitted by the licensee in support of the proposed TS change, supports the use of a single lower value for the DNBR penalty factor due to rod bowing. The single DNBR penalty factor would be included in the CPC and COLSS softwares. This would eliminate the need for determining the DNBR penalty factor for each batch of fuel assemblies based on its burnup and verifying the application of correct penalty factors in DNBR calculations.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of these standards by providing certain examples (48 FR 14870). One of the examples of actions involving no significant hazards considerations relates to a relief granted upon demonstration of acceptable operation from an operating restriction that was imposed because acceptable operation was not yet demonstrated.

The proposed change appears to be similar to the above example in that the CE study based on the ANO-2 irradiated fuel data appears to support acceptable operation of ANO-2 without the rod bow penalty factor surveillance requirement. Thus, the NRC staff proposes to determine that the proposed change involves no significant hazards consideration.

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

Attorney for licensee: Nicholas S. Reynolds, Esq., Bishop, Liberman, Cook, Purcell and Reynolds, 1200 Seventeenth Street, NW., Washington, D.C. 20036.

NRC Branch Chief: James R. Miller.

Arkansas Power and Light Company, Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: January 28, 1985.

Description of amendment request: The proposed amendment would revise Table 2.2-2, "CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS", to change the allowable ranges of the azimuthal tilt allowance (TR), the primary delta T calibration constant (TPC) and the neutron flux power calibration constant (KCAL). The proposed changes would make the values in the TS consistent with the present ranges of these addressable constants. The core protection calculators (CPC) addressable constants are provided to allow calibration of the CPC for more accurate indications of power level, RCS flow, and radial peaking. In addition, the CPC addressable constants allow inclusion of allowances for measurement uncertainties or inoperable equipment. The addressable constants are variables which are expected to be modified between cycles or even during reactor operation. By a Safety Evaluation dated June 19, 1981, the NRC staff approved a provision in the TS, i.e. TS 2.2.2, which allows the licensee to modify the addressable constants in accordance with the approved methodology and procedures to accommodate the fact that the addressable constants are expected to be modified even during reactor operation. The amendment request dated January 28, 1985 involves three additional issues which will be addressed in separate notices.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of standards for determining whether a proposed license amendment involves a significant hazards consideration by providing certain examples (48 FR 14870) of amendments not likely to involve significant hazards considerations. One of the examples relates to a change which either may result in some increase to the probability or consequences of a previously analyzed accident or may in some way reduce a margin of safety, but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan (SRP): For example a change resulting from the application of a small refinement of a previously used calculational model or design method.

It appears that the proposed changes are similar to the example cited in that they are refinements of the previously used calculational model for calibrating the CPC as a result of improved monitoring and additional operational experience.

On the basis of the above, the NRC staff proposes to determine that the requested actions involve no significant hazards consideration.

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

Attorney for licensee: Nicholas S. Reynolds, Esq., Bishop, Liberman, Cook, Purcell and Reynolds, 1200 Seventeenth Street, NW., Washington, D.C. 20036.

NRC Branch Chief: James R. Miller.

Arkansas Power and Light Company, Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: January 28, 1985.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) pertaining to the core protection calculators (CPC) addressable constants to accommodate the CPC functional modifications discussed in CEN-288(A) which was submitted by letter dated November 9, 1984. The CPC are an integral part of the ANO-2 reactor protection system (RPS). The addressable constants serve many CPC functions. Some addressable constants are provided to allow calibration of the CPC for more accurate indication of power level, reactor coolant flowrate and radial peaking. Other addressable constants allow inclusion of allowances for measurement uncertainties or inoperable equipment. The proposed changes replace one addressable constant and add two new addressable constants to Table 2.2-2 of the TS.

The amendment request dated January 28, 1985 involves three additional issues which will be addressed in separate notices.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of standards for determining whether a proposed license amendment involves a significant hazards consideration by providing certain examples (48 FR 14870) of amendments not likely to involve significant hazards considerations. One of the examples relates to a change which either may result in some increases to the probability of consequences of a previously analyzed accident or may in some way reduce a margin of safety, but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan (SRP): For example, a change resulting from the application of a small refinement of a previously used calculational model or design method. Each of the three proposed changes appears to be similar to the example cited and thus, the NRC staff proposes to determine that the proposed changes involve no significant hazards consideration. A description of each proposed change to the TS and a discussion of how each change is similar to the example cited are addressed below:

1. *Reactor Power Cutback Value Limit (RPCLIM)*—The proposed change would revise Table 2.2-2 of the TS to add the addressable constant RPCLIM (Point ID Number 103). The CPC algorithms which include RPCLIM are a part of a standard CPC software package update provided to the licensee by Combustion Engineering (CE), the CPC vendor. Even though ANO-2 does not contain the hardware necessary to implement reactor power cutback, the reactor power cutback algorithms will be included in the ANO-2 CPC update in order to reduce the differences with the CPC systems installed at other CE reactors. The effect of these algorithms on the ANO-2 CPC will be nullified through setting the applicable data base and addressable constant to zero.

The proposed change is similar to the example cited in that the change would provide for future refinement of the CPC by the addition of algorithms to support a reactor power cutback system. Furthermore, the proposed change will not increase the probability or consequences of a previously analyzed accident since the effect of the change would be deactivated by the use of appropriate addressable constant and data base.

2. *Secondary Calorimetric Power (PCALIB)*—The proposed change would revise Table 2.2-2 of the TS to add the addressable constant PCALIB (Point ID Number 104). The PCALIB is defined as calorimetric power at the time of the latest CPC thermal and neutron flux power calibration. This addressable constant would be added to one of the CPC algorithms which would apply a power dependent power measurement uncertainty. Under the current TS, a constant power measurement uncertainty for all power levels is applied in the CPC algorithms. The proposed change would result in the application of improved power measurement uncertainties since they vary with power levels. Thus, it appears that the proposed change is similar to the example cited in that it is a small refinement of the previously used calculational model. Further, the proposed change would enhance the RPS's ability to meet the criteria specified in SRP Section 7.2 "Reactor Trip System" in that it would enhance the CPC's ability to sense accident conditions and to initiate a reactor trip when appropriate.

3. *Temperature Shadowing Correction Factor Multiplier (CORR1)*—The proposed change would revise Table 2.2-2 of the TS to redefine the addressable constant CORR1 (Point ID Number 98). The addressable constant CORR1 is currently defined as "Temperature Shadowing Factor Correction Multiplier". Temperature Shadowing is the decalibration of ex-core neutron flux power resulting from changes in the reactor coolant density between the reactor core and the ex-core detectors. The proposed change would redefine the addressable constant CORR1 as "Reference Cold Leg Temperature" consistent with the CPC temperature shadowing algorithm modification and would reclassify it as a Type I addressable constant (Type I constants require periodic calibration). The CPC temperature shadowing algorithm modification which is discussed in CEN/288CA) would result in the temperature shadowing correction factor multiplier being redefined as a fixed constant in the CPC software.

The proposed change combine with the CPC temperature shadowing algorithm modification would provide a more accurate indication of power near the normal conditions and a more conservative temperature shadowing correction at conditions other than the normal conditions.

The proposed change appears to be similar to the example cited in that it is a refinement of a previously used calculational model for correcting ex-core detector signals for the effects of temperature shadowing. Furthermore, the proposed change would enhance the RPS's ability to meet the criteria specified in SRP Section 7.2, "Reactor Trip Systems" in that it would enhance the CPC's ability to sense accident conditions and to initiate a reactor trip when required.

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

Attorney for licensee: Nicholas S. Reynolds, Esq., Bishop, Liberman, Cook, Purcell and Reynolds, 1200 Seventeenth Street, NW., Washington, DC 20036.

NRC Branch Chief: James R. Miller.

Arkansas Power and Light Company, Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: January 28, 1985.

Description of amendment request: The amendment would revise the departure from Nucleate Boiling Ratio (DNBR) limit used by the Core Protection Calculators (CPC) to incorporate the findings of a recently completed Combustion Engineering (CE) study on rod bow penalty and penalties previously accounted for by one of the CPC addressable constants.

The DNBR is a unitless value calculated from reactor core thermal-hydraulic conditions on a real-time basis from an NRC approved empirical correlation. It is a measure of thermal margin. Maintaining core conditions such that DNBR is above a prescribed value ensures that the fuel cladding will not overheat during normal and abnormal plant operation. The CPC, which are an integral part of the reactor protection system (RPS) at ANO-2, monitor certain NSSS variables and initiate a reactor trip if fuel design limits are approached as a result of an abnormal event. The CPC addressable constants are provided to allow calibration of the CPC to more accurately predict reactor power levels and radial power peaking factors and to allow the CPC to account for measurement uncertainties or inoperable equipment.

In support of the revised rod bow DNBR penalty, the licensee has submitted a CE report, CEN-289(A). The CE report presents a refined calculational model based on accumulated irradiated fuel data specific to ANO-2. The present rod bow DNBR penalty is calculated based on extrapolation from a model for the 14×14 fuel design. The ANO-2 core contains 177 fuel assemblies of the 16×16 fuel design. The proposed DNBR limit would account for the new rod bow DNBR penalty.

In a Safety Evaluation (SE) dated July 21, 1981, the NRC staff approved a temporary adjustment on the BERR1 addressable constant using an NRC approved method to incorporate the difference between the NRC approved DNBR limit and the original CPC design DNBR limit. The proposed amendment would incorporate the penalties previously accounted for by the BERR1 addressable constant into the new DNBR limit.

As a result of the two adjustments on the DNBR limit discussed above, the DNBR limit would be revised from 1.24 to 1.25.

The proposed change on the DNBR limit is only one of four issues addressed in the application. The other issues will be the subject of separate notices.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the applications of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. One of the examples relates to a change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan (SRP); for example, a change resulting from the application of a small refinement of a previously used calculational model or design method.

It appears that the DNBR limit change emanating from the revised rod bow DNBR penalty is similar to the example cited in that the change results from the application of a small refinement of a previously used calculational model. The DNBR limit change to incorporate the penalties previously accounted by the BERR1 addressable constant represents an end to the use of a temporary adjustment procedure and incorporate the penalties which was found to be within all acceptable criteria with respect to the applicable SRP acceptance criteria (i.e., SRP Section 4.4) by the NRC staff.

Therefore, since the application for amendment involves a change similar to an example for which no significant hazards consideration exists, the NRC staff proposes to determine that the application for amendment involves no significant hazards consideration.

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

Attorney for licensee: Nicholas S. Reynolds, Esq., Bishop, Liberman, Cook, Purcell and Reynolds, 1200 Seventeenth Street, NW., Washington, D.C. 20036.

NRC Branch Chief: James R. Miller.

Baltimore Gas and Electric Company, Docket No. 50-317, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Calvert County, Maryland

Date of applications for amendment: December 31, 1984 and February 22, 1985.

Description of amendment request: The proposed amendment would change the Unit 1 Technical Specifications (TS) to reflect analyses performed in support of Cycle 8 operation.

Basis for proposed no significant hazards consideration determination: On April 6, 1983 the NRC published guidance in the **Federal Register** (48 FR 14870) concerning examples of amendments that are not likely to involve significant hazards considerations. One such example (iii) involves "For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable."

The proposed changes to the Unit 1 TS, submitted by applications dated December 31, 1984 and February 22, 1985 satisfy the criteria of example iii. Accordingly, the Commission proposes to determine that the proposed changes to the TS required for Unit 1 Cycle 8 operation involve no significant hazards considerations.

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

Attorney for licensee: George F. Trowbridge, Esq., Shaw, Pittman, Potts and Trowbridge, 1800 M Street, NW., Washington, D.C. 20036.

NRC Branch Chief: James R. Miller.

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 amendment request: January 31, 1985.

Description of amendment request: The proposed amendment would change the Unit 1 and Unit 2 Technical Specifications (TS) to reflect: (1) Changes to surveillance requirements for safety related hydraulic sway arrestors (snubbers) for Unit 1 only, (2) clarification of the degree of independence associated with the emergency core cooling system (ECCS) and shutdown cooling system, (3) deletion of a reactor vessel pressurization curve that is no longer needed for Unit 2 only, (4) a change to the containment isolation valve identification numbers, and (5) incorporation of the containment water level monitor including operability and surveillance requirements.

In reviewing the above proposed changes to the TS, we have determined that certain changes in the proposed TS are required. These changes were discussed with and agreed to by the licensee.

Basis for proposed no significant hazards consideration determination: The first TS change topic relates to the safety related hydraulic sway arrestors (snubbers) addressed in TS 3/4.7.8.1, "Snubbers". On April 19, 1984, the NRC issued Amendments 92 and 73 for Calvert Cliffs Units 1 and 2 which included a change to TS 3/4.7.8.1. This change allowed BG&E to replace snubbers with rigid supports (sway struts). During the Unit 1 Cycle 8 refueling outage, BG&E will replace a number of snubbers with sway struts as permitted by TS 3/4.7.8.1 and has proposed deletion of these snubbers from the TS. In addition, the licensee has requested a change to TS 3/4.7.8.1 to allow removal of three snubbers (1-11-12, 1-60-5, and 1-60-5A) without installing sway struts.

In both cases where the licensee has proposed removal of snubbers (with and without installation of a sway strut) stress calculation have been performed to demonstrate that no appreciable increase in seismic induced stress will occur in associated piping or equipment.

A second change associated with Unit 1 TS 3/4.7.8.1 involves the deletion of common-reservoirs notations from those designated snubbers in Unit 1 TS Table 3.7-4. These sixteen snubbers, associated with the Steam Generators, will be modified such that each snubber will have its own reservoir. The reserviors, together with all associated fittings, will be designed, manufactured, mounted and maintained to the same seismic standards as the snubbers which they serve. Removal of these common reservoirs and replacement with individual units improves the seismic design in that it eliminates the possibility that a single reservoir failure would result in eight snubbers being inoperable. Since these sixteen snubbers are the only snubbers served by common reservoirs, the surveillance requirements for these common reservoirs specified in TX 4.7.8.1f have been proposed for deletion. This proposed TS change was previously approved for the Unit 2 TS in License Amendment No. 73 which was issued on April 19, 1984.

The proposed changes in the snubbers addressed above and their associated TS assure an equivalent degree of seismic resistance; therefore, there will be no decrease in the seismic design margin. Accordingly, no increase in the probability of occurrence or consequences of seismic related failures will result. In addition, since only the seismic design of the facility is affected, no new or different kind of accident is likely to occur. For these reasons the Commission proposes to determine that the proposed changes to TS 3/4.7.8.1 involve no significant hazards considerations.

The licensee has proposed changes to TX 3.5.2, "ECCS Subsystems-T more than or equal to 300°F" and TS 3.9.8.2, "Shutdown Cooling and Coolant Circulation." Each of these TS requires that two "independent" subsystems (loops) of the respective systems be operable. The licensee has proposed deletion of the term "independent" as it applies to shutdown cooling and ECCS in TS 3.5.2 and 3.9.8.2, respectively.

The term "independent", when applied to system design, means that components have been arranged in subsystems which can function without interdependence. While both the ECCS and shutdown cooling systems contain major components which are arranged independently, both systems share common piping, within the respective system, and thus neither system is truly "independent." Deleting the word "independent" from TS 3.5.2 and 3.9.8.2 does not change the requirments of the TS. Both TS would still require that two subsystems (loops), at a minimum, be operable for each system. The term "independent" as used in TS 3.5.2 and 3.9.8.2 was used descriptively and, in these cases, incorrectly.

Based upon the above, we conclude that the proposed change would not involve an increase in the probability of occurrence or consequences of an accident previously evaluated. The two proposed changes are simply clarifications of Technical Specifications which more closely reflect actual plant design. The proposed TS change would not create the possibility of a new or different kind of accident from any accident previously analyzed. No new equipment, system alignments beyond those previously bounded by current Technical Specifications, or accident analyses are involved in the proposed change. Finally, the proposed change to the TS would not involve a significant reduction in the margin of safety. The TS are not being altered except to provide a clarification of actual plant design regarding the ECCS and shutdown cooling systems. Accordingly, the Commission proposes to determine that the proposed changes to TS 3.5.2 and 3.9.8.2 involve no significant hazard considerations.

The licensee has proposed deletion of TS Figure 3.4-2a, "Reactor Coolant System Pressure Temperature Limitations for 0 to 2 years of Full Power Operation." At the present time, TS Figure 3.4-2b, "Reactor Coolant System Pressure Temperature Limitations for 2 to 10 Years" provides the applicable limitations. Since Unit 2 has been in commercial operation for approximately seven years and has surpassed the two "effective full power years" point of reactor embrittlement, TS Figure 3.4-2a is no longer needed.

Deletion of TS Figure 3.4-2a in no way changes the applicable TS, prevents an error by removing information which is no longer applicable, and is thus administrative in nature. The Commission has provided guidance concerning the application of standards concerning "no significant hazards considerations" by providing certain examples (48 FR 14870). Purely administrative changes to Technical Specifications are explicitly considered not likely to involve significant hazards considerations. Accordingly, the Commission proposes to determine that the proposed deletion of TS Figure 3.4-2a involves no significant hazards considerations.

The licensee has proposed a change to TS Table 3.6-1, "Containment Isolation Valves." This table lists all containment isolation valves which are subject to operability and surveillance requirements. The licensee has proposed a change in the valve numbering system in TS Table 3.6-1 to achieve consistency with the operational piping and instrument diagrams (P&ID) and procedures used to perform the required surveillance on containment isolation valves.

The licensee has been involved in an effort to revise, upgrade, and standardize P&IDs. Associated with this effort they have performed a walkdown of all affected systems to verify the accuracy of affected drawings. TS Table 3.6.1 as presently written lists the valve designations used on contruction P&IDs. The proposed change would modify this table to reflect the numbers used on operational P&IDs. This would result in less chance of error while performing critical valve line-ups by making Table 3.6.1 consistent with operational procedures and P&IDs. The requested change is an administrative change and in no way changes existing operability or surveillance requirements in the TS. As previously indicated, administrative changes to the TS are not likely to involve significant hazards considerations. Accordingly, the Commission proposes to determine that the proposed change to TS Table 3.6.1 involves no significant hazards considerations.

The licensee has proposed the addition of containment water level monitor instrumentation to the operability and surveillance requirements of TS 3/4.3.3.6, "Postaccident Instrumentation."

On November 1, 1983, the NRC issued Generic Letter No. 83-37 (GL 83-37) to all pressurized water reactor licensees. This letter contained guidance concerning TS which the NRC believed to be appropriate as addressed in NUREG-0737, "Clarification of TMI Action Plan Requirements". The licensee responded, in part, to GL 83-37 via their applications for license amendments dated January 31, 1985. The licensee has proposed that existing TS Table 3.3-6, "Radiation Monitoring Instrumentation," and TS Table 4.3-3, "Radiation Monitoring Instrumentation Surveillance Requirements" would be modified to include Limiting Conditions for Operation (LCOs) and Surveillance Requirements for the containment water level monitor.

The proposed TS would increase the likelihood that the associated equipment will undergo appropriate surveillance and be available to assist in postaccident assessment. The proposed TS represents an additional limitation or restriction in that, in the event that the equipment becomes inoperable, the applicable LCO requires remedial action which was not previously required.

On April 6, 1983, the NRC published guidance in the Federal Register (48 FR 14870) concerning examples of amendments that are not likely to involve significant hazards consideration. One such example (ii) involves a change ". . . that constitutes an additional limitation, restriction, or control not presently included in the technical specifiication . . ." Since the proposed TS represent additional requirements not previously in the TS, these proposed changes are consistent with example (ii). Accordingly, the Commission proposes to determine that these proposed changes to the TS involve no significant hazards considerations.

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

Attorney for licensee: George F. Trowbridge, Esq., Shaw, Pittman, Potts and Trowbridge, 1800 M. Street, NW., Washington, D.C. 20036.

NRC Branch Chief: James R. Miller.

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

Date of amendment request: February 4, 1985.

Description of amendment request: The proposed amendment would change the Technical Specifications by imposing a new limit of 2 gpm increase, average over any 24-hour period, of reactor coolant leakage into the primary containment from unidentified sources. This limiting condition for operation (LCO) would apply only when the reactor has been in the RUN mode for more than 24 hours. More specific operational requirements are also proposed for the reactor coolant leakage detection system and the reactor pressure boundary leak detection system to account for the redundancy of these systems and the redundancy of components within subsystems.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning application of the standards for determining whether license amendments involve significant hazards considerations by providing certain examples (48 FR 14870). One example of an amendment that is considered not likely to involve a significant hazards consideration is "(ii) A change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications; for example, a more stringent surveillance requirement." The proposed amendment is similar to example (ii) since it would impose an additional limitation and more specific operational requirements. Based on this similarity, the staff has made a proposed determination that the application for amendment involves no significant hazards considerations.

Local Public Document Room location: Plymouth Public Library, North Street, Plymouth, Massachusetts 02360.

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

NRC Branch Chief: Domenic B. Vassallo.

Carolina Power & Light Company, Docket No. 50-325, Brunswick Steam Electric Plant, Unit 1, Brunswick County, North Carolina

Date of application for amendment: February 13, 1985.

Description of amendment request: The proposed Technical Specification changes request postponement of one full flow test of the core spray pumps until the primary containment suppression chamber is restored to its operational condition.

The licensee is presently planning to shutdown the Brunswick Steam Electric Plant, Unit 1 on or before March 31, 1985 for a 31 week outage (plus six weeks for contingencies) to refuel, perform maintenance work and modify the Mark I torus. In conjunction with the Mark I torus modifications, the suppression chamber will be drained and, therefore, it will not be possible to perform the full-flow surveillance test of the Core Spray System (CSS) wherein water is recirculated into the suppression pool.

This requirement will last be performed on approximately April 1, 1985. Due to the modifications being made to the suppression pool the maximum permissible interval between full flow tests will be exceeded before the next test. The licensee is, therefore, requesting a one time extension to the maximum surveillance interval during the upcoming refueling outage until within 48 hours after restoration of the suppression chamber to operable status, but in any case no later than October 30, 1985. Based on the present outage schedule, CP&L plans to restore the suppression chamber to operable status and perform Surveillance Requirement 4.5.3.1.c.1 by approimately August 29, 1985. This will extend the surveillance interval from the present maximum of 115 days to approximately 150 days. The October 30, 1985 date allows for contingencies in the completion of modification to the suppression pool making the total allowable surveillance interval 212 days.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance in the form of examples of amendments that are not considered likely to involve significant hazards considerations (48 FR 14870). The licensee's February 13, 1985 submittal included a discussion of the proposed action with respect to the no significant hazards consideration. The licensee also provided a discussion regarding the proposed Technical Specification (TS) change.

The licensee has determined and the NRC staff concurs that extending the surveillance interval, for a full flow test of the Core Spray System (CSS), from 92 days to a total allowable surveillance interval of 212 days does not constitute a significant reduction in the verification of operability or the availability of this system for the following reasons:

1. Normally, in the refueling operation (OPERATIONAL CONDITION 5), the CSS is not required to be operable, (and thus to have surveillance testing performed), if all of the following conditions are met: (1) The reactor vessel head is removed, (2) the refueling cavity is flooded, and (3) the spent fuel gates are removed. The CSS will be available for operation, if needed, during the relatively short interval when operability is required due to plant conditions (i.e., draining the refueling cavity until the suppression chamber is refilled).

2. The CSS consists of two independent subsystems, each with 100% capacity, thus providing redundant safety system subsystem.

3. Redundant systems that will be available to supply core reflood capability include the condensate system and the service water injection system, with a small volume available from the control rod drive system.

4. Surveillance is being performed every 12 hours to verify that the CSS has an operable water source (TS 4.5.3.1.a).

Surveillance is performed every 31 days to verify that the CSS is filled with water (TS 4.5.3.1.b.1).

Surveillance is performed every 31 days to verify that all valves in the CSS flow path are properly aligned (TS 4.5.3.1.b.2).

The proposed change pertaining to specification 4.5.3.1.c.1 represents a relaxation in the surveillance requirements. However, adequate precautions have been taken to ensure the availability of other means of cooling for the reactor core. Based on the foregoing discussion, the staff concludes that the results of this change, 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. Therefore, the Commission proposed to determine that these changes do not involve a significant hazards consideration.

Local Public Document Room location: Southport, Brunswick County Library, 109 W. Moore Street, Southport, North Carolina 28461.

Attorney for licensee: George F. Trowbridge, Esquire, Shaw, Pittman, Potts and Trowbridge, 1800 M Street, NW, Washington, D.C. 20036.

NRC Branch Chief: Domenic B. Vassallo.

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

Date of application for amendment: July 29, 1982 as supplemented August 30, 1984 and January 18, 1985.

Description of amendment request: The August 30, 1984 and January 18, 1985 submittals revise the July 29, 1982 submittal which was previously published in the Federal Register on August 23, 1983 (48 FR 38391). This amendment would modify the technical specifications to correctly identify certain relays associated with the plant emergency power supplies and provide correct set point values for actuating these relays.

Following investigation of a reactor scram, the licensee determined Degraded Voltage Surveillance Tests on Unit 1 were not being performed. The licensee's review of a previous modification revealed that incorrect relays were referenced in the plant modification and therefore, the incorrect set point values were incorporated in the technical specifications. Table 3.3.3-2, Item 5.a. describes Balance-of-Plant (BOP) busses 1C, 1D, 2C, and 2D for Device 27. The correct relay should have been Emergency Busses E-1, E-2, E-3, and E-4, Device 27/59E. The proposed changes to the technical specifications would correct this error and provide correct set point values for actuating the relays.

During the staff review of the proposed Technical Specifications certain clarifications were requested from the licensee. These clarifications were provided in letters dated August 30, 1984 and January 18, 1985.

The August 30, 1984 letter provided a revised voltage drop study. The results indicate that the distribution system remains above the aforementioned relays setpoint for the minimum grid voltage and the maximum plant load condition. Further, it demonstrates that the safety related loads will accelerate to full speed in less than the time delay specified in the TS. Thus, we find that the voltage profile for the BSEP Units 1 and 2 distribution system will remain above the relay trip curve for the minimum grid voltage.

The letter dated January 18, 1985 explained that the once per shift channel check performed on these relays consists of a check for relay targets which indicate if the relay is tripped or not and verification that the installed voltmeters on the 4.16 kV bus read greater than 3800 volts. This voltage information is also available (redundant) in the control room. Furthermore, the relays are arranged in a two-out-of-three logic for reliable actuation to avoid spurious trips.

The two supplemental letters provided additional information that further substantiated the proposed Technical Specifications change.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870). The examples involving no significant hazards consideration include "(i) a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature; and, (ii) a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications; for example, a more stringent surveillance requirement."

Example (i) encompasses the changes requested to correct the errors in identifying certain relays in the emergency power supplies. Example (ii) applies to the added requirements for these relays including proper set points, surveillance intervals and operability conditions. Therefore, since the application for amendment involves proposed changes that are similar to examples for which no significant hazards considerations exists, the Commission proposes to determine that the application for amendment involves no significant hazards consideration.

Local Public Document Room location: Southport, Brunswick County, Library, 109 W. Moore Street, Southport, North Carolina 28461.

Attorney for licensee: George F. Trowbridge, Esquire, Shaw, Pittman, Potts and Trowbridge, 1800 M Street, NW., Washington, D.C. 20036.

NRC Branch Chief: Domenic B. Vassallo.

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

Date of application for amendment: October 24, 1984, as supplemented February 27, 1985.

Description of amendment request: The proposed amendments would change the Limiting Condition for Operation (LCO), the Surveillance Requirements and the associated bases for Specification 3/4.6.1.3, Primary Containment Air Locks, to specifically address the air lock door interlocks. Additionally, the Technical Specifications will be reformatted to more closely follow the guidance of the NUREG-0123, Standard Technical Specifications.

The current Specification does not specifically address an inoperable door interlock in the LCO. As such, it could be interpreted that an inoperable door interlock falls outside the "degraded mode" permitted by Paragraph 3.6.1.3 (a) and (b). Were that to be the interpretation, this interlock would fall under Paragraph 3.6.1.3(c) which directs the plant to be in hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours. CP&L has concluded that this was not the intent of the Specification, since an inoperative door lock is clearly of a similar nature as the "degraded mode" permitted by paragraphs 3.6.1.3 (a) and (b).

The amendments, therefore, propose that the action described for an inoperable air lock door is sufficient to compensate for an inoperable door interlock.

The current Technical Specification requires that the operation of the air lock door interlock be verified every six months. This verification presents the following problems:

(1) The interlock surveillance is performed independently of the air lock operability requirements.

(2) The interlock surveillance cannot be performed when the unit is at power with the drywell inerted, as the drywell is inaccessible.

(3) A low power drywell entry just to perform the interlock surveillance would present an unnecessry safety hazard and increase radiation exposure to personnel performing the test.

The proposed revision requiring verification after each entry (except during periods of multiple entries where it is tested at least every 72 hours) will present the following resolutions:

(1) The interlock surveillance will be added to the air lock surveillance requirements. Thus, the two surveillances will be performed simultaneously, ensuring that the interlock is operable whenever the air lock is required to be operable.

(2) The surveillances will be performed with the unit in cold shutdown and prior to entering operational conditions 1, 2, or 3. The above surveillance requirement is in the Brunswick pre-startup checklist and in the drywell closure checklist. After the surveillance requirement is satisfactorily completed, access to the drywell is secured. This will ensure air lock and interlock operability in operational conditions 1, 2, or 3 and until another drywell entry is made. Whenever the drywell is entered, the surveillance requirement must be repeated prior to drywell closure.

(3) With the surveillance being performed simultaneously in cold shutdown, an additional drywell entry is not necessary. This will, therefore, reduce personnel exposure to radiation and prevent an additional safety hazard.

(4) The increased surveillance on the interlock will result in an increased level of confidence in the interlock's operability.

Additionally, the Specification is being reformatted to be consistent with NUREG-0123, the Standard Technical Specifications for General Electric Boiling Water Reactors.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870). The examples of actions involving no significant hazards consideration include: (i) A purely administrative change to the Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature; and (ii) a change that constitutes an additional limitation, restriction or control not presently included in the Technical Specifications.

The proposed change pertaining to the reformatting of the Specification is purely an administrative change as in example (i). The proposed revision requiring verification after air lock entry (except during periods of multiple entries where it will be tested at least every 72 hours) constitutes additional controls not presently included in the Technical Specifications and is, therefore, encompassed by example (ii). In addition, the change regarding the inoperable door interlock is also an additional control not presently included and, therefore, is encompassed by example (ii). Thus, the proposed changes discussed in this request are either administrative changes or constitute additional controls not presently included in the Specification and, therefore, conform to examples for which no significant hazards considerations exist.

Therefore, since the application for amendment involves proposed changes that are similar to examples for which no significant hazards considerations exist, the Commission proposes to determine that the proposed amendment involves no significant hazards considerations.

Local Public Document Room location: Southport, Brunswick County Library, 109 W. Moore Street, Southport, North Carolina 28461.

Attorney for licensee: George F. Trowbridge, Esquire, Shaw, Pittman, Potts and Trowbridge, 1800 M Street, NW., Washington, D.C. 20036.

NRC Branch Chief: Domenic B. Vassalo.

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

Date of application for amendment: October 29, 1984, as supplemented February 4, 1985.

Description of amendment request: The proposed amendments would change the Technical Specifications (TS) to incorporate the new reporting requirements as defined by the Commission in Generic Letter No. 83-43, dated December 19, 1983. In addition, recent organizational changes at Brunswick and various administrative changes are reflected in the proposed TS pages.

Section 50.72 of Title 10 of the Code of Federal Regulations has been revised and became effective January 1, 1984. A new § 50.73 of Title 10 of the Code of the Federal Regulations has been added and also became effective January 1, 1984. Section 50.72 revises the immediate notification requirements for operating nuclear power reactors. The new § 50.73 provides for a revised Licensee Event Report System.

Paragraph (g) of § 50.73 specifically states that: "the requirements contained in this section replace all existing requirements for licensees to report 'Reportable Occurrences' as defined in individual plant Technical Specifications." The definition "Reportable Occurrence" will be replaced by a new term, "Reportable Event." These changes will be made in the current version of Standard Technical Specifications (STS) for all nuclear power reactors and in the Technical Specifications for plants not yet licensed.

The changes relating to the revised reporting requirements are in accordance with 10 CFR 50.72 and 10 CFR 50.73 and with the guidance provided by the Commission in Generic Letter 83-43 and are made at the Commission's request. In addition, organizational changes are proposed which included: (1) Inclusion of Brunswick personnel title and organizational changes; (2) correction of typographical errors; (3) clarification of terms and mathematical symbols; and (4) repagination. The organizational changes consist of: deletion of the office of Manager—Plant Operations; addition of the Manager—Outages and his staff; a title change from Director—Planning and Scheduling to Manager—Site Planning and Contral; a title change from Manager—Operations QA/QC to Manager—QA/QC Brunswick and Robinson; a title change from Principal QA Specialist Performance Evaluation Unit to Manager—QA Services; and a shift of responsibility for Fire Brigade training from the Manager—Operations to the Director—Training.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of its standards set forth in 10 CFR 50.92 for no significant hazards consideration by providing certain examples published in the Federal Register on April 6, 1983 (48 FR 14870). Examples of an amendment likely to involve no significant hazards consideration include: (i) A change which is purely administrative in nature, for example, a correction of an error, or a change in nomenclature; and (vii) a change to make a license conform to changes in the regulations, where the license change results in very minor changes to facility operations clearly in keeping with the regulations.

The staff has reviewed the proposed amendment and finds that the revisions relating to the new reporting requirements fall under the criteria of example (vii) since they are clarifying requirements made by a change in the regulations and made at the request of the Commission. The typographical clarification and repagination changes are found to be similar to the administrative changes in the cited example and therefore fall under example (i). The organizational changes do not involve (1) a significant increase in the probability or consequences of an accident previously evaluated, (2) the possibility of a new or different kind of accident from any accident previously evaluated, nor (3) a significant reduction in a margin of safety. On this basis, the Commission proposes to determine that these proposed amendments do not involve a significant hazards consideration.

Local Public Document Room location: Southport, Brunswick County Library, 109 W. Moore Street, Southport, North Carolina 28461.

Attorney for licensee: George F. Trowbridge, Esquire, Shaw, Pittman, Potts and Trowbridge, 1800 M Street, NW., Washington, D.C. 20036.

NRC Branch Chief: Domenic B. Vassallo.

Commonwealth Edison Company, Docket Nos. 50-373 & 50-374, La Salle County Station, Units 1 & 2, La Salle County, Illinois

Date of amendment request: January 15, 1985.

Description of amendment request: The proposed amendments to Operating License NPF-11 and Operating License NPF-18 would revise the La Salle, Units 1 and 2 Technical Specification 4.6.5.3.d.3 to change the method for calculating the kilowatt capacity of Standby Gas Treatment Heaters when they are tested and to reflect the 3 kW higher capacity of newly installed heaters. The changes are required because current Technical Specification, which does not account for the bus voltage in the kilowatt capacity calculation is ambiguous, and because new heaters have been installed.

The duct heaters for the Standby Gas Treatment System are designed to reduce the relative humidity of the airflow to a maximum of 70% relative humidity at the worst inlet conditions. The heaters originally installed to provide this function had a nominal rating of 20 kW (at 480 volts). The purpose of Technical Specification 4.6.5.3.d.3 surveillance requirement is to ensure that the heaters perform their function without major degradation. The present method of testing the performance of the heaters is based on a ±2 kW acceptance range for the previous 20 kW heaters without reference to the bus voltage during testing.

Recently, the heaters were replaced with ones having a slightly higher heat rating of 23 kW (at 480 bus voltage). Additionally, due to variations in actual bus voltage at the time of test, the allowable kW for each heater should be compared to the amount of voltage supplied, based on the textbook relationship $W = V^2/R$, where W is the resulting electrical heat developed by the heater, V is the applied voltage and R is the resistance of the heater for more accurate calculation during tests. The proposed amendment would change the performance criteria of the heaters to account for their capacity based on test at a nominal 23 kW capacity. The diesel generator loading which powers the heaters will not be significantly affected by the nominal 3 kW per heater increase, and the small increase in heater downstream temperature due to the increased kilowatt will not affect the thermal safety setting of 220 °F in the heaters. The change in the method of calculating the heater capacity will provide more accurate test information on the heaters function.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of standards for a no significant hazards consideration determination by providing certain examples (48 FR 14879). One of the examples (vi) of actions involving no significant hazards considerations is a change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria. These proposed amendments simply clarify the Tech Specs to indicate that allowable heater capacity is voltage dependent and to reflect the higher capacity of new heaters, and do not change the intent of the Technical Specifications nor permit testing or operation outside acceptable criteria. Therefore, since the application for amendments involve proposed changes that are similar to an example for which no significant hazards consideration exists, the staff has made a proposed determination that the application for amendments involves no significant hazards consideration.

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

Attorney for licensee: Isham, Lincoln and Burke, Suite 840, 1120 Connecticut Avenue, NW., Washington, D.C. 20036.

NRC Branch Chief: A. Schwencer.

Commonwealth Edison Company, Docket Nos. 50-373 & 50-374, La Salle County Station, Units 1 & 2, La Salle County, Illinois

Date of amendment request: February 21, 1985.

Description of amendment request: The proposed amendments to Operating License NPF-11 and Operating License NPF-18 would revise the La Salle, Units 1 and 2 Technical Specifications in Tables 4.3.1.1-1, 4.3.2.1-1, 4.3.3.1-1 and 4.3.5.1-1 to delete the channel check requirements from certain instruments. These instruments contain Barton differential indicating switches to measure vessel level and various system flows. These switches are installed in the Reactor Protection Systems, Primary Containment Isolation Systems, Emergency Core Cooling Systems, and Reactor Core Isolation Cooling Actuation Systems.

These Barton differential pressure indicating switches have not met the qualification requirements of 10 CFR 50.49 and are being replaced by qualified differential pressure switches manufactured by Static-O-Ring, Inc. These new switches are blind differential switches and do not have local indication. During the preparation of the La Salle Units 1 & 2 Technical Specifications, required channel checks were added where indication was available for performing these checks. Since these pressure switches are now being upgraded to meet 10 CFR 50.49, these channel checks are not possible and must be deleted from the Technical Specifications. It should be noted, though, while these specific instrument channels are deleted, in all cases except one, other instrumentation from the same reactor vessel reference and variable legs are still required to have channel checks.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility 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; (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 license has determined and the NRC staff agrees that the proposed amendments will not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated because this change only removes the channel check requirements. Channel functional testing and calibrations are still periodically required to ensure system availability as necessary. Single failure criteria is not affected by this revision.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated because failure of these instruments is evaluated and no new accident is postulated from removing the channel check requirement.

(3) Involve a significant reduction in the margin of safety because the availability of safety related systems is not significantly affected.

Accordingly, the Commission proposes to determine that the proposed changes to the Technical Specifications involve no significant hazards considerations.

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

Attorney for licensee: Isham, Lincoln and Burke, Suite 840, 1120 Connecticut, Avenue, NW., Washington, D.C. 20036.

NRC Branch Chief: A. Schwencer.

Commonwealth Edison Company, Docket No. 50-265, Quad Cities Nuclear Power Station, Unit 2, Rock Island County, Illinois

Date of amendment request: January 3, 1985.

Description of amendment request: The proposed amendment would change the Technical Specifications for the reactor scram system. The change would provide new limiting conditions for operation and surveillance requirements for a newly modified scram system having improved reliability. The modifications were implemented per an NRC Order issued on June 24, 1983. The proposed changes to the Technical Specifications are based upon the licensee's final design of its scram system and its review of model technical specifications provided as guidance by the NRC staff.

Basis for proposed no significant hazards consideration determination: The licensee submittal of January 3, 1985, contained an evaluation of the proposed action, and a proposed no significant hazards consideration determination, based on the following considerations.

Subsequent to a failure of 76 of 185 control rods to fully insert at Browns Ferry Unit 3 in response to a manual scram signal, the Commission had embarked on an indepth review of the BWR control rod drive system which identified a number of design issues requiring both short and long term corrective measures. On October 1, 1980 letters were sent to all BWR licensees requesting commitments to reevaluate the present scram system and modifying it as necessary to meet both the design and performance criteria as developed by the BWR Owners Subgroup. Accordingly, a confirmatory order was written June 24, 1983 for Quad Cities Unit 2 regarding a schedule for implementation of the long term corrective actions. That Confirmatory Order also provided model technical specification changes. Based on Commonwealth Edison's final design and upon a review of the model technical specifications, Commonwealth Edison is proposing a number of changes to Appendix A of the Technical Specification for Quad Cities Unit 2 in accordance with the formentioned Confirmatory Order.

The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing specific examples. The examples of actions involving no significant hazards consideration include: (ii) Changes that constitute an additional limitation or restriction or control not presently within the technical specifications e.g., a more stringent surveillance requirement.

The changes proposed in this application for amendment are encompassed by this example because of the additional limitations and restrictions that will be added by this Technical Specification amendment.

Therefore, since the application for amendment involves a proposed change that is similar to an example for which no significant hazards consideration exists, Commonwealth Edison has made a proposed determination that the application involves no significant hazards consideration.

The staff has reviewed the licensee's significant hazards consideration determination and, based on this review, the staff has made a proposed determination that the application for amendment involves no significant hazards considerations.

Local Public Document Room location: Moline Public Library, 504—17th Street, Moline, Illinois 61265.

Attorney for licensee: Mr. Robert G. Fitzgibbons, Jr., Isham, Lincoln, & Beale, Three First National Plaza, Suite 5200, Chicago, Illinois 60602.

NRC Branch Chief: Domenic B. Vassallo.

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

Date of application for amendments: October 19, 1984, augmented by letters dated December 20, 1984 and February 14, 1985.

Description of amendments request: The amendment would change (a) the hot channel factor limits and (b) limiting conditions for operation of the accumulator system. Both changes result from the revised Emergency Core Cooling System (ECCS) analysis.

Basis for proposed no significant hazards consideration determination: The revised ECCS analysis resulted in new values for peak cladding temperature, total core hydrogen generation and local cladding oxidation. All new values are well within the respective limits set forth in 10 CFR 50.46. The new analysis demonstrates increased safety margins using previously approved analysis models and methods which are in compliance with 10 CFR 50.46 and Appendix K. In addition, the results also meet the criteria set forth in Section 15.6.5, Loss of Coolant Accident of the Standard Review Plan. The Commission has provided guidance concerning the application of these standards by providing certain examples (48 FR 14870). The examples of actions involving no significant hazards include actions which may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: For example, a change resulting from the application of a small refinement of a previously used calculational model or design method. The changes requested fall in this category. On the above basis, the staff proposes to conclude that the amendments involve a no significant hazards consideration.

Local Public Document Room location: Zion-Benton Library District, 2800 Emmaus Avenue, Zion, Illinois 60099.

Attorney to licensee: P. Steptoe, Esq., Isham, Lincoln and Beale, Counselors at Law, Three First National Plaza, 51st Floor, Chicago, Illinois 60602.

NRC Branch Chief: Steven A. Varga.

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

Date of application for amendments: February 5, 1985.

Description of amendments request: The amendments would revise the actions required in the event of an inoperable rod due to a rod urgent failure condition. The amendments would reduce challenges to plant safety systems as requested in Inspection Report Nos. 50-295/81-09 and 50-304/81-05.

Basis for proposed no significant hazards consideration determination: The existing Technical Specifications require if more than one control rod is inoperable, except due to a rod urgent failure, the reactor must be shutdown within four hours. For inoperability due to rod urgent failure, if the affected assemblies cannot be returned to service within two hours, the reactor shall be shutdown within 4 hours. The amendment being proposed would provide that, for inoperability caused by a rod urgent failure condition, if the affected assemblies cannot be returned to service within twenty-four hours the reactor shall then be shutdown within the next four hours.

A rod urgent failure indicates equipment failure in the rod control system power or logic cabinets. The rod urgent failure condition will inhibit the rod control system's ability to move rods, but will not affect the ability of the control rods to be tripped.

The Commission's example of actions involving no significant hazards considerations (48 FR 14870) include: "(vi) A change which may either result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but when the results of the change are clearly within all acceptable criteria with respect to the system or component. . .". The above example fits the proposed change. The staff, therefore, proposes to conclude that the proposed changes to the Technical Specifications involve no significant hazards consideration.

Local Public Document Room location: Zion-Benton Library District, 2600 Emmaus Avenue, Zion, Illinois 60099.

Attorney to licensee: P. Steptoe, Esq., Isham, Lincoln and Beale, Counselors at Law, Three First National Plaza, 51st Floor, Chicago, Illinois 60602:

NRC Branch Chief: Steven A. Varga.

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

Date of amendment request: November 19, 1984.

Description of amendment request: The proposed amendment would: (1) Incorporate additional technical specifications for the new Control Room Emergency Air Cleanup Systems and (2) add fire detectors, sprinklers, and a hose station to the Tables of required fire protection equipment that have been added to the facility.

Basis for proposed no significant hazards consideratioan determination: The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870, April 6, 1983). One of the examples (ii) of actions not likely to involve a significant hazards consideration relates to a change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications. The proposed changes (1) and (2) above add such limitations and controls for equipment presently not included in the technical specifications.

Therefore, because this amendment request involves only changes of the type specified in example (ii) of the Commission's guidance, the staff proposes to determine that the proposed changes would not involve a significant hazards consideration.

Local Public Document Room location: Kalamazoo Public Library, 315 South Rose Street, Kalamazoo, Michigan 49007.

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

NRC Branch Chief: John A. Zwolinski.

Dairyland Power Cooperative, Docket No. 50–409, La Crosse Boiling Water Reactor, Vernon County, Wisconsin

Date of amendment request: October 18, 1984, as revised January 10, 1985.

Description of amendment request: The existing Provisional Operating License (DPR–45) for the La Crosse Boiling Water Reactor has license conditions which prohibit the license from possession or use of more than 100 millicuries each of any by-product material for sample analysis or instrument calibration, or 100 mililigrams each of any source or special nuclear material for sample analysis or instrument calibration, with the exception of up 10 curies of cesium-137 which may be used in the form of a sealed source for instrument calibration. On October 18, 1984, Dairyland Power Cooperative (DPC) proposed a change to the conditions of the facility operating license to allow the receipt, possession, and use of more than 10 curies of cesium-137 in the form of sealed sources for instrument calibration. Subsequently, on January 10, 1985, DPC revised the earlier submittal and requested that all quantity limitations on possession of by-product or special nuclear material for sample analysis or instrument calibration be removed from the facility opearting license. These changes would make the La Crosse license conditions for by-product materials consistent with those in licenses currently being isued by the NRC to new plants which allow the possessin of these materials "in amounts as required."

Basis for proposed no significant hazards consideration determination: The licensee's requests to eliminate limitations on the amount of certain redionuclides at La Crosse would allow the possession and use of a more accurate device for calibration of various radiation monitoring instruments throughout the plant. The proposed change would make the La Crosse license consistent with the conditions now incorporated in operating licenses issued by the NRC to new plants. The intent of the proposed change is to allow the licensee to have greater flexibility in selection of radioactive sources for calibrating radiation detection equipment. Failure of radioactive sources has extremely low consequences for members of the general public and thus is not considered in licensing evaluations of nuclear plants. Although it would allow an increase in the amount of radioactive material used at La Crosse, the proposed change is not expected to significantly increase the amount used at the site for the identified purposes. The licensee's technical specifications for handling and control of this material are consistent with the Standard Technical Specifications which are implemented at new plants and are consistent with current licensing criteria. Therefore, based upon all of the above, the staff concludes that the proposed change does not involve a significant hazards consideration determination since it: (1) Does not involve a significant increase in the probability or consequences of a previously evaluated accident, (2) does not create the possibility of a new or different kind of accident from an accident previously evaluated, and (3) does not involve a significant reduction in a margin of safety.

Local Public Document Room location: La Crosse Public Library, 800 Main Street, La Crosse, Wisconsin 54601.

Attorney for licensee: O.S. Heistand, Jr., Esquire, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, D.C. 20036.

NRC Branch Chief: John A. Zwolinski, Branch Chief.

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

Date of amendment request: October 1, 1984.

Description of amendment request: The amendments would modify the Environmental Technical Specifications (Appendix B) to delete the requirement for aerial photography which has been employed to determine the effects of cooling tower drift on the surrounding environment.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). One of the examples of actions involving no significant hazards consideration is example (iv), a relief granted upon demonstration of acceptable operation from an operating restriction that was imposed because acceptable operation was not yet demonstrated. This assumes that the operating restriction and the criteria to be applied to a request for relief have been established in a prior review and that it is justified in a satisfactory way that the criteria have been met.

The proposed amendments constitute a change to grant relief upon demonstration of acceptable operation. The aerial photography program requirement was instituted due to the unknown effect of the deposition of cooling tower drift upon the environment prior to plant operation. The program has been successfully completed with no adverse environmental impact. Therefore, this change is similar to example (iv). On this basis, the Commission proposes to determine that the amendment request involves no significant hazards considerations.

cal Public Document Room
ion: Appling County Public Library,
City Hall Drive, Baxley, Georgia.
orney for licensee: G.F.
bridge, Shaw, Pittman, Potts and
bridge, 1800 M Street, NW.,
ington, D.C. 20036.
C Branch Chief: John F. Stolz.

gia Power Company, Oglethorpe r Corporation, Municipal Electric ority of Georgia, City of Dalton, gia, Dockets Nos. 50-321 and 50- dwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

e of amendment request: nber 7, 1984, as corrected nber 20, 1985.

cription of amendment request: mendments would modify the Unit 1 Technical Specifications

elete the requirement for istration diesel generator oility when core spray stystems, essure core injection systems , plant service water systems or actor heat removal (RHR) systems operable.

dd a requirement to verify offsite availability and correct breaker ents.

place the monthly diesel tor test with a test schedule ment based on the number of s during the previous 100 valid

ld a requirement that during the st the diesel accelerates to nuous speed within 12 seconds reases the minimum load for strating operability.

minate the requirement for nal operability testing of the very 24 hours following the esting.

crease the time allowed to an inoperable diesel to operable rom 7 days to 3 days.

d a requirement for an annual n the number of tests and for each diesel.

mendments would modify the nit 2 Technical Specifications

ease the time allowed to restore wo inoperable diesel generators erable status from 2 hours to 24

lace the requirements for testing t 3, 7, 14, or 31 day intervals the total number of failures out st 100 valid tests of all diesels at th only two of these test , 7 and 14 days, based on the of failures out of the last 100 ts of all diesels at a unit with of these test intervals, 7 and 14 sed on the number of failures out of the last 100 valid tests for the specific diesel being tested.

3. Revise the 18-month cycle, 24-hour diesel test requirement to require that the overload test be performed during the last two rather than the first two hours of the test.

4. Extend the test interval for verifying operability of diesel air start receivers from 18 months to 5 years.

5. Replace the requirements to report failures for each diesel test and to provide a supplemental report if more than seven failures occurred during the last 100 tests with an annual report like the one discussed above for Hatch Unit 1 (Item 7).

The amendments would modify the Technical Specifications for both Hatch Units 1 and 2 to:

1. Add a once a year 7-day inoperability exception for each individual diesel and two 18-day inoperability exceptions for all the diesels in a unit to the 3-day inoperable limit for an inidividual diesel.

2. Increase the time allowed for a diesel to accept full load during a test from 2 minutes to 5 minutes.

3. Increase the time allowed to verify that a diesel is operable after declaring an offsite power source component of another diesel to be inoperable from "immediately" to 24 hours, except that incease of loss of an offsite power source component, the diesel will not have to be tested if it has been successfully tested within the previous 7 days.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance for the application of the criteria in 10 CFR 50.92 by providing examples of amendments that are considered not likely to inivolve a significant hazards consideration (48 FR 14870). One such example is (ii), a change that constitutes an additional limitation, restriction or control not presently included in the Technical Specifications.

Items 2, 3, 4, 6 and 7 listed above as changes to the Hatch Unit 1 Technical Specifications are similar to this example.

Another such example (i) of action not likely to involve significant hazards consideration is a purely administrative change to the Technical Specifications. Item 5 listed above as a change to the Hatch Unit 2 Technical Specifications is similar to this example.

The Commission has also provided standards for determining whether a significant hazards consideration exists [10 CFR 50.92(c)]. A proposed amendment to an operating license for a facility 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.

The Commission's staff, in Generic Letter (GL) 84-15 ("Proposed Staff Action to Improve and Maintain Diesel Generator Reliability"—July 2, 1984), indicated that requirements for testing diesel generators while emergency core cooling equipment is inoperable results in excessive testing and increased degradation of diesel engines. The staff recommended, therefore, that these testing requirements be deleted from the Technical Specifications. Item 1 listed above as a change to Hatch Unit 1 Technical Specifications is one such item as addressed in GL 84-15. The licensee stated in its November 7, 1984, letter that the above proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because it will eliminate a practice of unnecessary and abusive diesel generator testing which can contribute to accelerated diesel generator wear, which consequently degrades diesel generator reliability and availability. The licensee stated that this change will not create the possibility of a new or different kind of accident from any accident previously evaluated because no physical modifications are required to be made to the plant and performance of onsite emergency power systems as described in the Final Safety Analysis Report (FSAR) will remain unchanged. The licensee also stated that this change does not involve a significant reduction in a margin of safety because with the proposed change failures of the core spray, LPCI or RHR service water system components will not adversely affect the reliability and performance of the diesel generators.

The Commission's staff agrees with the licensee's evaluation in this regard, and accordingly, the staff proposes to find that this change involves no significant hazards considerations.

Item 5 listed above as a change to Hatch Unit 1 Technical Specifications, Items 1 through 4 above for Hatch Unit 2 Technical Specifications, and Items 1 through 3 above for both Hatch Units 1 and 2 Technical Specifications are changes directed at enhancing the reliability of the diesel generators. GL 84-15 expressed the staff's position that frequency of fast start tests from ambient conditions of diesel generators falls within example (vii). Because these amendments fall within examples of actions not likely to involve significant hazards considerations, the staff proposes to determine that the requested action involves no significant hazards consideration.

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

Attorney for licensee: G.F. Trowbridge, Esquire, Shaw, Pittman, Potts, and Trowbridge, 1800 M Street, NW., Washington, D.C. 20036.

NRC Branch Chief: John A. Zwolinski.

Indiana and Michigan Electric Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan

Date of amendment request: February 14, 1985.

Description of amendment request: The amendments would revise the Technical Specifications by updating the plant heatup and cooldown curves to reflect the recent reactor vessel material surveillance capsule examination and analysis.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards for determining whether license amendments involve no significant hazards considerations by providing certain examples (48 FR 14871). One of these examples (ii) is a change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications. The proposed amendments are directly related to this example in that revised heatup and cooldown curves are required to meet the reactor vessel fracture toughness requirements in 10 CFR Part 50, Appendix G. These new limits on heatup and cooldown constitute an additional limitation, restriction, and control not presently included in the current heatup and cooldown curves in the Technical Specifications. On this basis, the Commission proposes to determine that the amendments involve no significant hazards consideration.

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

Attorney for licensee: Gerald Charnoff, Esquire, Shaw, Pittman, Potts and Trowbride, 1800 M Street, NW., Washington, D.C. 20036.

NRC Branch Chief: Steven A. Varga.

Iowa Electric Light and Power Company, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: November 9, 1984, as revised January 18, 1985.

Description of amendment request: The proposed amendment would change the Duane Arnold Energy Center (DAEC) Technical Specifications to incorporate changes to fire protection surveillance requirements taking into account the installation of fire detection systems in most areas of the plant containment safety-related equipment.

Presently the DAEC Technical Specifications require that a continuous fire watch must be established within one hour in the event a fire barrier is found to be nonfunctional. The recent installation of a fire detection system at DAEC, would permit a relaxation of current continuous fire watch requirement. The licensee has therefore requested to change that requirement to an hourly watch in the event a fire barrier is found to be nonfunctional. The licensee has also proposed an additional change to correct a typographical error in the Technical Specifications.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of standards of no significant hazards consideration determination by giving certain examples (48 FR 14870, April 6, 1983). One of the examples of actions considered likely to involve no significant hazards consideration is example (vi) relating to a change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan. The proposed change would relax the continuous fire watch requirement to an hourly fire watch in the presence of a fire detection system. Such a relaxation may result in some reduction in a safety margin, but the results of the change in conjunction with the fire detection system modifications are clearly within all acceptable criteria in the Standard Review Plan Section 9.5.1.

Another example of actions involving no significant hazards consideration is example (i) a purely administrative change to Technical Specifications: For example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or change in nomenclature. The proposed change to correct an error is encompassed by the above cited examples.

Therefore, since the application for amendment involves proposed changes similar to examples for which no significant hazards consideration exists, the staff has made a proposed determination that the application involves no significant hazards consideration.

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

Attorney for licensee: Jack Newman, Esquire, Harold F. Reis, Esquire, Newman and Holtzinger, 1025 Connecticut Avenue, NW., Washington, D.C. 20036.

NRC Branch Chief: Domenic B. Vassallo.

Iowa Electric Light and Power Company, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: December 7, 1984.

Description of amendment request: The Iowa Electric Light and Power Company (the licensee) proposes to revise the Duane Arnold Energy Center (DAEC) Technical Specifications to reflect conformance to the Type C testing criteria of 10 CFR Part 50, Appendix J, Paragraph III.C.2.(b) for containment isolation valves.

The present Technical Specifications distinguish between those valves tested with air and those tested with water. The DAEC Technical Specifications state that valves tested with water shall be pressurized to 54 psig (P_a). However, 10 CFR Part 50, Appendix J, Paragraph III.C.2.(b) requires the tests be conducted at a pressure not less than 1.10 P_a. The proposed change request would revise the Technical Specifications to conform to the Type C testing requirements of 10 CFR Part 50, Appendix J, Paragraph III.C.2.(b), and would change the test pressure from 54 psig to the more restrictive pressure of 1.10 P_a.

Basis for no significant hazards consideration determination: The Commission has provided guidance concerning the application of standards of no significant hazards consideration determination by providing certain examples (48 FR 14870, April 6, 1983). One such example (ii) relates to a change that constitutes an additional limiting restriction, or control not presently included in the Technical Specifications: For example, a more stringent surveillance requirement. Changing the Type C testing pressure for containment isolation valves from the current pressure of P_a to a pressure of 1.10 P_a is an additional restriction not currently in the DAEC Technical Specifications. The proposed change is, therefore, encompassed by the above cited example.

Therefore, the staff has made a proposed determination that the application involves no significant hazards consideration.

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

Attorney for licensee: Jack Newman, Esquire, Harold F. Reis, Esquire, Newman and Holtzinger, 1025 Connecticut Avenue, NW., Washington, D.C. 20036.

NRC Branch Chief: Domenic B. Vassallo.

Iowa Electric Light and Power Company, Docket No. 50-331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: January 11, 1985, as supplemented March 15, 1985.

Description of amendment request: The proposed amendment would revise the Duane Arnold Energy Center (DAEC) Technical Specifications (TS) reflecting the previously proposed operation with an extended load line limit, and the presently proposed improvements to the Average Power Range Monitor (APRM) and Rod Block Monitor (RBM). The APRM, RBM and TS (ARTS) improvements are intended to increase the plant operating efficiency, update the compliance with the thermal margins requirements, improve the accuracy and response of the pertinent instrumentation, and to improve the man/machine interface. The proposed improvement program would modify the Technical Specifications as follows:

1. The RBM setpoints will be changed from flow biased to power-dependent settings; and
2. The APRM system flow-biased setpoints setdown requirement will be eliminated, and the flow-biased APRM setpoint values will be changed.

Basis for proposed no significant hazards consideration determination: The proposed submittal is similar to the previous requests for amendments to the Hatch and the Monticello plants. In those cases, the staff made a determination that a significant hazards consideration was involved. The staff's determination was based on the novelty and the complexity of the ARTS improvement program at that time. Subsequent reviews of the Hatch and the Monticello applications have clarified the safety issues associated with the ARTS program, so that a no siginificant hazards consideration finding can now be made.

The Commission has provided guidance concerning determination if significant hazards consideration exists, by providing certain standards (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility 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.

1. *APRM System Changes.* Adjusting the flow-biased equations of APRM scram and rod blocks will allow operation above the 100% load line at less then rated power/flow conditions. Operation above the 100% load line is achieved by withdrawal of control rods at low power/flow conditions using preestablished withdrawal sequences. The only accidents initiated by withdrawal of control rods are the Rod Withdrawal Error and Control Rod Drop Accidents, which require a withdrawal of a rod out-of-sequence and a decoupling of the control blade from its drive, respectively. Both of these events are independent of the withdrawal sequence used or final rod pattern chosen. Thus, the probability of these events is not increased from that analyzed in the Updated Final Safety Analysis Report (UFSAR) by operation above the 100% load line. The operation above the 100% load line is bounded by the analyses conducted at the 100% power and 100% flow conditions, except for the Feedwater Controller Failure transient. However, the Feedwater Controller Failure is not the most limiting transient for determining the operating limits. Thus, the consequences of an accident are not increased from those previously analyzed in the UFSAR.

The requirement to setdown the APRM flow-biased scram and rod block equations when the maximum fraction of limiting power density (MFLPD) exceeds the fraction of rated core thermal power (FRP) is being eliminated. In order to insure that the consequences of any abnormal operating transient or loss-of-coolant accident (LOCA) are not increased by this change, flow and power-dependent Minimum Critical Power Ratio (MCPR) and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits are being added to the Technical Specifications and are based upon the supporting transient and LOCA analyses at off-rated conditions. Removing the thermal peaking factor setdown requirement will not increase the probability of any such transient or accident, as it is not the initiating event of any accident.

Deleting the APRM flow-biased rod block as a Limiting Safety System Setting (LSSS) will not change the probability of any accidents because no credit is taken for its function in any safety evaluation.

Adjusting the APRM flow-biased scram and rod blocks to allow operation above the 100% load line impacts only the Rod Withdrawal Error and Control Rod Drop Accidents. Both the accidents are analyzed in the UFSAR. Therefore, the possibility of a different type of accident is not created.

The thermal peaking factor setdown requirement was instituted to protect the fuel from transients and accidents initiated from off-rated conditions. Replacing the peaking factor setdown function with equivalent flow and power-dependent MCPRs and MAPLHGRs will not create a new or different type of accident.

Deleting the APRM Rod Block as an LSSS will not introduce a new or different accident because no credit is taken for its function.

The supporting analyses of design basis events and abnormal operating transients were conducted above the 100% load line (100% Power, 87% Flow). In all cases, except the Feedwater Controller Failure (FWCF) transients, the results were bounded by those analyzed at the 100% power, 100% flow condition and thus the margin of safety for these events are not reduced above those previously analyzed. While the results of the FWCF transient at the (100, 87) point were slightly worse than those at rated conditions, the margin of safety is not degraded as this is a nonlimiting event and is not used for determining the operating limit MCPRs, which define the margin of safety.

Replacing the requirement to perform the thermal peaking factor setdown with equivalent flow and power-dependent MCPRs and MAPLHGRs will ensure that the margin of safety is not reduced by eliminating the setdown requirement.

Deleting the APRM Rod Block as an LSSS will not reduce the margin of safety as the APRM Rod Block will be maintained in the other sections of the Technical Specifications.

2. *Rod Block Monitor (RBM) System Changes.* The current flow-biased RBM rod blocks protect localized regions of the core from inadvertent withdrawals of control rods which would violate the

Safety Limit MCPR, i.e., rod withdrawal error (RWE) accidents. The new power-referenced RBM setpoints are based upon the supporting RWE analyses, which ensure with a 95% probability at a 95% confidence level (95/95 limits) that the Safety Limit MCPR will not be violated by an single withdrawal of a control rod. Therefore, replacing the current flow-biased rod blocks with power-referenced rod blocks will not increase the probability of any event (RWE) previously analyzed. In addition, the probability of an RWE event is not increased by this change due to the increased operator confidence in the new RBM system, as the new hardware for the RBM system, which supports the power-dependent limits, is simpler and more reliable than the old system.

The definition of a Limiting Control Rod Pattern (LCRP) has been revised based upon the supporting RWE analyses provided in the licensee's submittal. These analyses show, with 95/95 limits, that an RWE initiated when the plant is not on an LCRP will not violate the Safety Limit MCPR with the RBM bypassed. Therefore, revising the operability and surveillance requirements of the RBM system based upon the new definition of an LCRP will not increase the probability or magnitude of the consequences of any accident previously analyzed.

The RBM system's design function is to prevent localized fuel failures due to RWE accidents, which have previously been analyzed. That design function has not been changed by converting to power-dependent setpoints and defining new operability and surveillance requirements of the RBM system based upon the new definition of an LCRP. Therefore, the possibility of a new or different accident is not created.

The supporting analyses of RWE accidents demonstrate that the MCPR margin of safety is not reduced by the new power-dependent setpoints and operability and surveillance requirements for the RBM system. Therefore, the proposed change will not result in any reduction of a safety margin.

Therefore, since the application for amendment involves proposed changes which meet the Commission's standards for cases where no significant hazards consideration exists, the staff has made a proposed determination that the application involves no significant hazards consideration.

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

Attorney for licensee: Jack Newman, Esquire, Harold F. Reis, Esquire, Newman and Holtzinger, 1025 Connecticut Avenue, NW., Washington, D.C. 20036.

NRC Branch Chief: Domenic B. Vassallo.

Iowa Electric Light and Power Company, Docket No. 50-331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: January 11, 1985.

Description of amendment request: The proposed amendment would change the Duane Arnold Energy Center (DAEC) Technical Specifications to incorporate revisions in pressure-temperature operating limits for the reactor vessel and revise the minimum temperature, for which the reactor vessel head bolting studs can be in tension, from 100 °F to 74 °F. The pressure-temperature operating limits for the reactor vessel are being revised to reflect minor changes in the fracture toughness due to 6 effective full power years of neutron fluence on the vessel. The revision of the bolting stud tension temperature from 100 °F to 74 °F is shown by the licensee's analyses to comply with the Commission's regulation 10 CFR 50 Appendix G.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of standards of no significant hazards determination by giving examples (48 FR 14870, April 6, 1983). One such example of actions considered likely to involve no significant hazards consideration is example (vi) relating to a change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan. . .: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method.

The pressure-temperature fracture toughness has been revised due to 6 full power years of neutron fluence. The revision incorporates a calculational refinement accounting for neutron fluence. The reduction of the temperature from 100 °F to 74 °F for which reactor vessel head bolting studs can be in tension will comply with the Commission's regulations. Both of the above changes may result in reduction of safety margins, but the results of these changes are clearly within the acceptable criteria with respect to components specified in the Standard Review Plan section 5.3.1.

Therefore, since the application for amendment involves proposed changes similiar to an example for which no significant hazards consideration exists, the staff has made a proposed determination that the application invoves no significant hazards consideration.

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

Attorney for licensee: Jack Newman, Esquire, Harold F. Reis, Esquire, Newman and Holtzinger, 1025 Connecticut Avenue, NW., Washington, D.C 20036.

NRC Branch Chief: Domenic B. Vassollo.

Nebraska Public Power District, Docket No. 50-298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: December 20, 1984, as supplemented by submittal dated February 22, 1985.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) for (1) average power range monitor (APRM) flow transmitter calibration, (2) definition of "OPERABLE-OPERABILITY", (3) NUREG-0737, Item II.K.3.18, "ADS Logic Modification", and (4) Cooper Nuclear Station organization change.

(1) *APRM Flow Transmitter Calibration.* The proposed change would correct Section 4.1, Bases and a note to Table 4.2.C, "Surveillance Requirements for Rod Withdrawal Block Instrumentation" to reflect actual conditions resulting during calibration of the APRM Flow Biasing Network. Current TS state that during calibration ". . .a zero flow signal will be sent to half of the APRM's resulting in a half scram and rod block condition." However, each reactor recirculation flow unit, when in the calibration mode, actually sends a full flow signal to half of the APRM's producing a rod block but not a half scram. The licensee states that even without the half-scram signal, a substantial margin from fuel damage is provided by the 120% high flux scram which is in effect during calibration.

(2) *Definition of "OPERABLE-OPERABILITY".* The proposed change would expand the present definition of operability to explicitly include functionality of support systems and components such as instrumentation, control, power and other auxiliary systems. The present definition of operable requires only that a system or component be capable of performing its

intended function in its required manner in order to be considered operable.

(3) *NUREG–0737, Item II.K.3.18, "ADS Logic Modifications"*. The proposed change would revise the TS relative to the automatic depressurization system (ADS) to be consistent with ADS actuation logic modifications previously approved by the staff in a letter dated June 6, 1984. The approved ADS logic modifications result in eliminating the need for manual actuation for transient and accident events which do not directly produce a high drywell pressure signal. Consequently, the licensee will eliminate the high drywell pressure permissive from the ADS logic and add a manual inhibit switch. The proposed TS change would delete all references to 2 psig drywell pressure from Table 3.2.B for ADS circuitry requirements and add surveillance requirements for the manual inhibit switches to Table 4.2.B.

(4) *Cooper Nuclear Station Organization Change.* The proposed TS change reflects changes in the CNS organization as follows: (a) Changes of position title, (b) a change in the reporting requirements for the Chemistry and Health Physics (HP) Supervisor, and (c) the addition of a control room supervisor to the plant staff. The changes of position title are proposed for the convenience of the licensee to more accurately describe position duties and responsibilities. The proposed change in the reporting requirements of the Chemistry and HP Supervisor were made to address the guidelines of Regulatory Guide 1.8 for experience level. As a result of these comments, it is proposed that a broken line be added to the organization chart between the Senior Rad/Tech Advisor and the Chemistry and HP Supervisor to indicate the oversight function of the Senior Rad/Tech Advisor. This latter manager will review and direct the efforts of the Chemistry and HP Supervisor until he has attained the requisite experience level specified in Regulatory Guide 1.8. The addition of the control room supervisor, who will be a qualified senior reactor operator, reflects the changes in staffing that were made in accordance with the minimum staffing requirements of 10 CFR 50.54.

Basis for proposed no significant hazards consideration determination:

(1) *APRM Flow Transmitter Calibration.* The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). One of the examples of actions involving no significant hazards considerations is example (i) purely administrative changes to the Technical Specifications, for example to achieve consistency throughout the Technical Specifications, correction of an error or a change in nomenclature. The change proposed by the licensee relative to the APRM flow transmitter calibration involves only a correction to the Bases section and to an explanatory note to a table to reflect actual plant design. These changes do not affect any surveillance operations of limiting conditions for operation. The proposed changes are descriptive and purely administrative in nature. As such, the proposed changes fall within the scope of example (i). On this basis, the Commission proposes to determine that these changes involve no significant hazards considerations.

(2) *Definition of "OPERABLE-OPERABILITY".* The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). One of the examples of actions involving no significant hazards considerations, i.e. example (ii), relates to a change that constitutes an additional limitation, restriction or control not presently in the Technical Specifications. The proposed change would make the definition of operable more limiting and is, therefore, similar to this example. The Commission therefore proposed to determine that this action involves no significant hazards considerations.

(3) *NUREG–0737, Item II.K.3.18, "ADS Logic Modifications".* The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license to a facility 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.

The proposed TS changes result from plant modifications previously approved by the staff in a letter dated June 6, 1984. The licensee has evaluated the proposed TS changes against each of the above three criteria and has provided the following results of the evaluation in the application dated December 20, 1984:

1. Because the proposed change eliminates the need for operator actuation of the ADS System for certain transient and accident events, it decreases the consequences of an accident previously evaluated and has no effect on its probability of occurring.

2. Because the proposed change does not introduce any new mode of operation, the possibility of an accident of a different type than analyzed in the Final Safety Analysis Report would not result from the change; therefore, the proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Because the proposed amendment eliminates the need for operator actuation of ADS for certain events, thereby freeing the operator to monitor and evaluate the accident or transient and take actions to combat any additional concerns, the proposed change does not reduce, but enhance the margin to safety.

The staff agrees with the licensee's evaluation that the proposed change meets the three criteria of the Commission's guidance as stated above. On this basis, the Commission proposes to determine that the application does not involve a significant hazards consideration.

(4) *Cooper Nuclear Station Organization Change.* The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). Examples of actions not likely to involve significant hazard consideration include actions specified as (i) purely administrative changes to the Technical Specifications, (ii) changes that constitute an additional limitation, restriction, or control not presently include in the Technical Specifications, and (vii) a change to make a license conform to changes in the regulations, where the license change results in very minor changes to facility operations clearly in keeping with the regulations. The proposed changes in position title are administrative in nature and fall within example (i) above. The proposed change relative to the reporting requirements of the Chemistry and Health Physics Supervisor reflects an additional control on this position and as such is similar to example (ii). The addition of a control room supervisor who is a qualified senior reactor operator was made to comply with the requirements of 10 CFR 50.54. This proposed change is therefore similar to example (vii). On this basis, the Commission proposes to determine that these changes involve no significant hazards considerations.

Local Public Document Room location: Auburn Public Library, 118 15th Street, Auburn, Nebraska 68305.

Attorney for licensee: Mr. G.D. Watson, Nebraska Public Power

District, Post Office Box 499, Columbus, Nebraska 68601.

NRC Branch Chief: Domenic B. Vassallo.

Nebraska Public Power District, Docket No. 50-298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: January 10, 1985, as supplemented by submittal dated February 28, 1985.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to support operation of Cooper Nuclear Station (CNS) during the upcoming fuel Cycle 10 and to expand the flexibility of plant limits to permit operation with barrier-type fuel and hafnium (General Electric Hybid I) control rods. The proposed amendment would revise the following areas:

1. Rod Block Monitor (RBM) Upscale Trip Setting.
2. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Curves.
3. Minimum Critical Power Ratio (MCPR) Curves.
4. Description of Control Rod Materials.

These proposed changes are described in more detail below.

1. *RBM Upscale Trip Setting.* The proposed amendment would change the way in which the RBM upscale trip setting is expressed in the TS. The existing trip setting is expressed in terms of recirculation loop flow plus a constant value. The proposed amendment would replace the constant with a variable parameter that would vary with core configuration. For each change in core configuration the value of the variable would be calculated using the methodology delineated in the latest NRC-approved version of General Electric (GE) Report "General Electric Standard Application for Reactor Fuel", (NEDE-24011-P-A). The methodology is identical to that currently used to determine the value of the trip setting as it is now expressed in the TS.

2. *MAPLHGR Curves.* The proposed amendment would revise the title of the existing MAPLHGR curves to indicate the applicability of the curves for use with barrier-type fuel as well as with the currently used fuel at CNS. The addition of barrier fuel to these curves does not necessitate a change to the numerical values represented by the curves. Barrier-type fuel is similar to previously-used fuel at CNS except that a thin Zirconium liner is added to the inner surface of the cladding to reduce cladding failures due to pellet-clad interaction. The barrier fuel design has been incorporated into the current revision of GE Report, "General Electric Standard Application for Reactor Fuel", (NEDE-24011-P-A-6, April 1983) and has been determined by the NRC to be acceptable.

3. *MCPR Curves.* The proposed amendment would result in small changes to the MCPR curves to be consistent with the operation of the reactor core during fuel Cycle 10. In addition, the range of applicability would be changed so that separate curves would be used for beginning of cycle and end of cycle instead of a single curve for the entire cycle. The MCPR curves have been calculated using the NRC-approved methodology of NEDE-24011-P-A-6, April 1983. The Cycle 10 reload will use fuel identical to that used previously at CNS.

4. *Description of Control Rod Materials.* The proposed amendment would change the description of control rod materials in TS Section 5.2 to include the use of hybid design hafnium control rod assemblies. The Hybrid I Control Rod (HICR) Assembly has been designed by General Electric (GE) to be used as direct replacement for the present control rod assemblies. The original control rods contain only boron carbide, B_4C, as the absorbing material. The HICR assemblies use B_4C absorber cubes and three solid hafnium rods in the outside edge of each wing. The HICR design will lengthen control rod lifetime.

The description of these control rods was submitted to the NRC by GE in topical report NEDE-22290. Based on the staff's evaluation of the information provided in (a) NEDE-22290, (b) a meeting with GE representatives, and (c) responses to NRC staff questions, the staff concluded that there is reasonable assurance that the substitution of Type I HICRs for other approved GE control blades will not result in unacceptable hazards to the public and should, in fact, result in improved control blade performance and a positive contribution to reactor safety. Therefore, NEDE-22290, as amended to incorporate the staff's safety evaluation, was approved as a referenced document for the GE Type I HICR by NRC letter dated August 22, 1983.

Basis for proposed no significant hazards consideration determination:

1. *RBM Upscale Trip Setting.* The Commision has provided standards for determining whether a significant hazards consideration exists [10 CFR 50.92(c)]. A proposed amendment to an operating license for a facility involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or conequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in margin of safety.

The proposed amendment would base the calculation of the trip setting on NRC-approved methodology but would not specify a value for setting as is presently the case. The NRC staff has determined that the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated because the value of this trip setting would not change for a particular core configuration, just the way it is expressed in the TS. Only small changes in trip setting are expected to result from changes in core configuration. In either case, the resultant trip setting is based on the same NRC-approved calculational methodology.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated because the resultant value of trip setting would be based on NRC-approved calculational methods as described above.

(3) Involve a significant reduction in margin of safety because, as discussed above, only small changes in trip setting are expected to result from changes in core configuration and for a particular core configuration there would be no difference in trip setting as a result of the amendment.

Based on the above, the staff proposes to find that the proposed TS changes to the BRM upscale trip setting do not involve a significant hazards consideration.

2. *MAPLHGR Curves.* The Commission has provided standards for determining whether a significant hazards consideration exists [10 CFR 50.92(c)]. A proposed amendment to an operating license for a facility involves no significent hazards considerations 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; (2) create the possibility of a new or different kind of accident previously evaluated; or (3) involve a signifiant reduction in margin of safety.

The proposed amendment would change the title of the MAPLHGR curves to indicate their applicability with barrier-type fuel, but would not change the numerical values of the curves. The lack of change in the MAPLHGR curves indicates that the type of barrier fuel involved in this proposed change has the same nuclear and thermal characteristics as previously-used fuel. The MAPLHGR values for the barrier fuel were determined using NRC-approved methodology. Because of these considerations, and because the barrier-type fuel design has been found to be acceptable to the NRC, we conclude that the possibility or consequences of a previously evaluated accident would not be significantly increased and the possibility of new accidents would not be created. Also, because the design of the barrier fuel is intended to reduce the possibility of fuel cladding failure, we conclude that the margin of safety would be increased. Based on these conclusions, we find that the proposed changes to the MAPLHGR curves meet the Commission's standards cited above, and the staff, therefore, proposes to determine that this change to the TS does not involve a significant hazards consideration.

3. *MCPR Curves*. The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). One of the examples of actions involving no significant hazards considerations, i.e. example (iii), is a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility are involved. The proposed changes to the MCPR curves are necessitated by a core reload at CNS with fuel identical to types previously used at the facility. The proposed changes to the MCPR values were calculated using NRC-approved methodology. Therefore, the staff finds that the proposed changes fall within the Commission's example (iii) of an action not likely to involve a significant hazards consideration.

Accordingly, the staff proposes to determine that the proposed change to the MCPR curves involves no significant hazards considerations.

4. *Description of Control Rod Materials*. The Commission has provided standards for determining whether a significant hazard consideration exists [10 CFR 50.92(c)]. A proposed amendment to an operating license for a facility 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.

The staff has reviewed the proposed amendment and the related topical report. The licensee concludes that the proposed amendment does not involve a significant hazards consideration and based on the following discussion the staff concurs with this conclusion.

The material evaluation, which includes the chemical, physical, mechanical and irradiation properties, indicates that data and experience demonstrate acceptable corrosion resistance in high temperature water and steam exist for hafnium in BWR control rods. The physical properties expected to be germane to control application indicate acceptable performance in the BWR environment.

The mechanical evaluation indicates that the thermal expansion and irradiation growth of hafnium will not interfere with handle and velocity limiter.

A nuclear evaluation indicates that the HICR will have no significant impact on core and fuel operation when used as a replacement for the current B_4C control rod assemblies. Experiments provide critical benchmarks for calculations and illustrate a minimum impact on local power and flux distributions with all hafnium rods. An even smaller impact is expected for HICR which is a mixture of hafnium and B_4C. Therefore, the HICR can be used without change in the current lattice physics treatment of control rod assemblies and current design procedures.

Thermal-hydraulic evaluation shows that the maximum temperature of the new rods is not significantly different from the currently used control rod assemblies.

An accident evaluation shows that the HICR weight and envelope are identical to the current assemblies. The mechanical and nuclear properties of the HICR do not differ from the current assemblies in any measures that might be significant during normal or accident conditions.

The HICR is, except for minor differences, mechanically identical to the BWR assemblies for which many reactor years of safe operating experience are available. Accordingly, the mechanical safety analysis for the HICR is enveloped by the mechanical safety analyses for the current assemblies.

The reactor core response for the HICR design has been evaluated against the current control rod design for comparison with linear heat generation, minimum critical power ratio and maximum average planar heat generation limits. The HICR weight and rod worth are the same as the current control rod design, therefore the scram speed and scram reactivity are the same and the above limits are not affected by the change.

Based on the above, the staff has determined that: (1) The probability of occurrence or the consequences of an accident would not be increased above those anlayzed in the Final Safety Analysis Report (FSAR) because the weight and envelope of the HICR are identical to those of the currently used assemblies, and the nuclear and mechanical properties of the HICR do not differ from currently used assemblies in a significant way; (2) the possibility of an accident different from those analyzed in the FSAR would not result from these changes because, in addition to the above, these systems would not be operated in a manner new or different from that described in the FSAR; and (3) the margin of safety as analyzed in Technical Specifications would not be reduced because the proposed amendment involves no significant relaxation of the criteria used to establish safety limits, no significant relaxation of the bases for limiting safety system settings, and no significant relaxation in limiting conditions for operation. Therefore the staff finds 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; (2) create the possibility of a new or different kind of accident; or (3) involve a significant reduction in a margin of safety. Accordingly, the Commission proposes to determine that the proposed changes to the TS involve no significant hazards considerations.

Local Public Document Room location: Auburn Public Library, 118 15th Street, Auburn, Nebraska 68305.

Attorney for licensee: Mr. G.D. Watson, Nebraska Public Power District, Post Office Box 499, Columbus, Nebraska 68601.

NRC Branch Chief: Domenic B. Vassallo.

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

Date of amendment request: February 6, 1985.

Description of amendment request: The proposed changes in the Technical Specifications are based on anticipated Cycle 7 fuel inventory. Although not expected, additional Technical

Specification changes may be necessary depending upon the final inventory. During the Cycle 6 shutdown, plans have been made to replace all fuel assemblies that contain leakers. Due to the uncertainty in the fuel inventory, the final Cycle 7 reload design cannot be performed until after the Cycle 6 shutdown and a determination is made as to the exact fuel inventory to be used in Cycle 7 (expected mid-May 1985).

The proposed changes in the Technical Specifications modify the allowable region of operation when the core power distribution is monitored by the Excore Detector Monitoring System. The new curve allows a wider range of operation (i.e., higher thermal power and a larger axial shape index). The change establishes two curves to be used. A new curve for the allowable thermal power vs. axial shape index has been developed for the case where the total radial peaking factor (F_{xy}) is less than or equal to 1.62 while the present curve in the Technical Specifications will be applicable for F_{xy} values less than or equal to 1.719.

The proposed changes trade range in radial peaking for more range in axial shape index. The maximum radial peak is specified in the Technical Specifications as a limit on F_{xy} while the maximum axial peak is specified by limits on the axial shape index. The new curve will still assure that the peak linear heat rate (i.e. 15.6 Kw/ft) assumed in the LOCA analysis is not exceeded. The increase in the allowable value of the axial shape index will be offset by a decrease in the allowable value F_{xy} without changing the design basis value for linear heat rate. Consequently, all safety analyses involving linear heat rate are not impacted by the change. Similarly, the transients for which Departure from Nucleate Boiling Ratio (DNBR) is a concern are unaffected by the change, since the proposed curve is still bounded by the curve in Technical Specification 3.26 (Figure 3.2-4). Figure 3.2-4 provided the shapes that were input into all DNBR design basis analyses.

Basis for proposed no significant hazards consideration determination: Based on the above information, we conclude that the proposed Technical Specification change would not impact the previously derived maximum allowable linear heat rate or other parameters which could adversely impact plant transient or accident analyses. Therefore, the proposed change 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. Accordingly, the staff proposes to determine that the proposed change does not involve a significant hazards consideration.

Local Public Document Room location: Waterford Public Library, Rope Ferry Road, Route 156, Waterford, Connecticut.

Attorney for licensee: Gerald Garfield, Esq., Day, Berry and Howard, One Constitution Plaza, Hartford, Connecticut 06103.

NBC Branch Chief: James R. Miller.

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

Date of amendment request: February 7, 1985.

Description of amendment request: The amendment would add new technical specifications addressing the operability and surveillance requirements for the New Toxic Gas Monitoring System. A new Toxic Gas Monitoring System was recently installed and declared operational. The amendment would also make administrative changes to correct the duplication of numbering of a table. The current technical specifications have two different tables numbered 2-8.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). One of the examples (ii) of actions not likely to involve a significant hazards consideration relates to changes that constitute additional restrictions or controls not presently included in the technical specifications. Another one of the examples (i) of an action not likely to involve a significant hazards consideration is a purely administrative change to the technical specifications. The proposal to add new technical specifications to address the operability and surveillance requirements for the new Toxic Gas Monitoring System comes under example (ii). The proposal to renumber one table to avoid number duplication comes under example (i). Based upon the above, the staff proposes to determine that the application does not involve a significant hazards consideration.

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

Attorney for licensee: Leboeuf, Lamb, Leiby, and MacRae, 1333 New Hampshire Avenue, NW., Washington, D.C. 20036.

NRC Branch Chief: James R. Miller.

Pacific Gas and Electric Company (PG&E), Docket No. 50-133, Humboldt Bay Nuclear Power Plant, Unit No. 3, Humboldt County, California

Date of amendment request: July 30, 1984.

Description of amendment request: PG&E proposed: (1) To amend License No. DPR-7 to possess-but-not-operate status; (2) to delete license conditions related to seismic modifications, investigations and analysis required prior to NRC authorization of a return to power operation; (3) to revise the Technical Specifications (TSs) to reflect the possess-but-not-operate Status of license; and (4) to decommission Humboldt Bay Unit No. 3 in accordance with a decommissioning plan submitted with the application. Items 2, 3 and 4 above will be noticed separately in the Federal Register. This notice applies to item (1).

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards for a no significant hazards consideration determination by providing certain examples (April 6, 1983, 48 FR 14870). Example (ii) is a change that constitutes an additional limitation, restriction, or control not presently included in the TSs. The proposed action to amend License No. DPR-7 to possess-but-not-operate status is a more restrictive license than the present license because the present license permits operation of the facility if certain conditions related to seismic concerns are met. Therefore, since the change is encompassed by example (ii) of the Commission's guidance, the staff proposes to determine that the proposed action does not involve a significant hazards consideration.

Local Public Document Room location: Eureka Humboldt County Library, 421 I Street (County Courthouse), Eureka, California 95501.

Attorney for licensee: Phillip A. Crane, Jr., Pacific Gas and Electric Company, Post Office Box 7442, San Francisco, CA 94120.

NRC Branch Chief: John A. Zwolinski.

Pennsylvania Power & Light Company, Docket No. 50-387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of amendment request: October 30, 1984 as supplemented February 20, 1985.

Description of amendment request: The proposed amendment would delete license condition 2.C.(4)(b) to Facility Operating License No. NPF-14 for Susquehanna Steam Electric Station (SSES), Unit 1. License Condition 2.C.(4)(b) requires that the licensee provide a new stability analysis indicating the results for appropriate exposure core conditions prior to startup following the first refueling outage.

In the Code of Federal Regulations (10 CFR Part 50 Appendix A), General Design Criteria (GDC) 12, "Suppression of Reactor Power Oscilations," states:

The reactor core and associated coolant, control, and protectin systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

The licensee has stated that the license condition is more restrictive than GDC 12 in that it does not include the option to detect and suppress power oscillations. With the approval of Technical Specifications implementing the guidance of General Electric Service Information Letter No. 380 (GE-SIL-380), Revision 1, the licensee has stated that procedures for detecting and suppressing power oscillations at SSES have been implemented, and compliance with the intent of license condition 2.C.(4)(b) has been satisfied.

Basis for proposed no significant hazards consideration determination: The method to detect and suppress power oscillations incorporated into SSES Technical Specifications, based on guidance contained in GE-SIL-380, would satisfy GDC 12. Since the purpose of license condition 2.C.(4)(b) is to assure that GDC 12 is satisfied, and since the license has an alternate and acceptable means to satisfy GDC 12 (i.e., the detection and suppression of power oscillations), there is no longer any purpose served by license condition 2.C.(4)(b). Because the purpose of license condition 2.C.(4)(b) has been satisfied by other means, deletion of this license condition will not significantly increase the probability or consequence of accidents previously evaluated, wuill not create the possibility of a new and different accident from any previously evaluated, and will not significantly reduce a safety margin. On that basis the NRC staff proposes to find this proposed change does not involve a 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 & Trowbridge, 1800 M Street NW., Washington, D.C. 20036.

NRC Branch Chief: A Schwencer.

Pennsylvania Power & Light Company, Docket No. 50-387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of amendment request: January 15, 1985 as supplemented February 21, 1985.

Description of amendment request: The proposed amendent would revise the Unit 1 Technical Specifications (TS) to support the operation of Susquehanna Steam Electric Station (SSES), Unit 1 at full rated power during the upcoming Cycle 2. The proposed amendment request to support this reload, changes the Technical Specifications in the following areas: (1) Establishes operating limits for all fuel types for the upcoming Cycle 2 operation; (2) establishes the Average Power Range Monitor setpoints; (3) reflects the replacement of approximately one quarter of the core with Exxon fuel assemblies for the upcoming Cycle 2 operation; and (4) modifies the bases section to account for the use of Exxon fuel assemblies.

To support the license amendment request for operation of Susquehanna Unit 1 during Cycle 2 the licensee submitted as attachments to the application the following:

I. Susquehanna SES Unit 1 Cycle 2 Reload Summary Report (NPE-84-015)
II. Susquehanna Unit 1 Cycle 2 Reload Analysis (XN-NF-84-116)
III. Susquehanna Unit 1 Cycle 2 Plant Transient Analysis (XN-NF-84-118)
IV. Susquehanna Unit 1 Cycle 2 Plant Transient Analysis Recirculation Pump Run-Up Results (XN-NF-84-118 Supplement 1)
V. Susquehanna Unit 1 LOCA-ECCS Anlysis MAPLHGR Results (XN-NF-84-119)
VI. Susquehanna SES Unit 1 Cycle 2 Proposed Startup Physis Tests Summary Description

During the first refueling outage 192 General Electric (GE) initial fuel assemblies (approximately one quarter of the core) will be replaced with new but substantially similar Exxon, type XN-1, fuel assemblies.

Basis for Proposed No Significant Hazards Consideration Determination: The proposed amendent to the Susquehanna Technical Specification to support this reload is very similar to Example (iii) provided by the Commission of the types of amendments not likely to involve significant hazards consideration. Example (iii) is an amendment to reflect a core reload where:

(1) No fuel assemblies significantly different from those found previously acceptable to the Commission for a previous core at the facility in question are involved;

(2) No significant changes are made to the acceptance criteria for the Technical Specifications;

(3) The analytical methods used to demonstrate conformance with the Technical Specifications and regulations are not significantly changed; and

(4) The NRC has previously found such methods acceptable.

This reload will consist of 764 assemblies, 572 of which are once burned GE fuel assemblies and 192 of which are new ENC, type XN-1 fuel assemblies. The Exxon fuel assemblies are very similar to the GE fuel assemblies except for slight differences in the mechanical, thermal-hydraulic and nuclear design.

Although the Exxon fuel is very similar to the GE fuel, the slight differences in mechanical, thermal-hydraulic, and nuclear design of the bundles, and the use of different analysis methodologies, required that a wide range of reanalyses be performed by Exxon Nuclear Company (ENC). This included reanalyzing for anticipated operations occurrences, performing LOCA and MAPLHGR analyses for the Exxon fuel, and analyzing for the rapid drop of a high worth control rod to assure that excessive energy will not be deposited in the fuel. Analyses for normal operation of the reactor consisted of fuel evaluations in the areas of mechanical, thermal-hydraulic and nuclear design.

The use of the ENC type XN-1, fuel assemblies and the associated analytical methods used for the Cycle 2 reload analyses have been previously approved by the Commission's staff for use in other boiling water reactors (BWR's). Based on previous experience, the staff has determined that only small differences result between the use of Exxon or GE analytical methods.

The other difference between the Cycle 1 core and the Cycle 2 core reload is in the core loading pattern. Cycle 1 is a standard GE BWR/4 initial core configuration consisting of fuel assemblies of similar enrichments placed in a specific zone within the core. In contrast the Cycle 2 core will be based on the conventional scatter load principle where fresh reload assemblies are scatter loaded throughout the core except for the center region and the core periphery. Changing from a zone core loading pattern used during first fuel cycle to a scatter loading pattern for the new reload assemblies during the second cycle is an accepted reload method that has been approved by the staff for other BWR plant reloads.

Thus, this core reload involves the use of fuel assemblies that are not significantly different from those found previously acceptable to the Commission for a previously core at this facility. The request for amendment changes the TSs to reflect new operating limits associated with the fuel to be inserted into the core are based on the new core physics and are within the acceptance criteria.

In the analyses supporting this reload, there have been no significant changes in acceptance criteria for the Technical Specifications, and those analytical methods used have previously been found acceptable.

The only difference between this reload and Example (iii) provided by the Commission is related to the use of the Exxon analytical methods which are different than those used for Cycle 1. However the Exxon analytical methods have been previously approved by the staff for use in the BWR's and the analytical results are not significantly different from those found previously acceptable to the Commission for the initial core at the facility.

On the bases of the similarity between the proposed amendment and the Commissions Example (iii) and the fact that the analytical methods used have been previously approved by the staff and do not provide results significantly different, the Commission's staff has concluded that operation of the facility in accordance with the proposed reload amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any previously evaluated; or (3) involve a significant reduction in the margin of safety. Based on the foregoing discussion, the Commission's staff proposes to determine that the amendment request does not involve a 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 & Trowbridge, 1800 M Street NW., Washington, D.C. 20036.

NRC Branch Chief: A Schwencer.

Pennsylvania Power & Light Company, Docket No. 50-387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of amendment request: January 31, 1985.

Description of amendment request: NUREG-0737 Item II.K3.18 required modification of the automatic depressurization system (ADS) actuation logic to eliminate the need for manual actuation to assure adequate core cooling. The addition of the ADS drywell pressure bypass timer and manual inhabit switch to the Susquehanna Steam Electric Station (SSES), Unit 1, satifies this requirement. Since the ADS is considered one of the safety systems used to assure emergency core cooling, license condition 2.C.(28)(e) required that the licensee propose Technical Specifications to cover the equipment installed that eliminated the need for manual actuation of the ADS. The changes to the Technical Specifications requested in this amendment, except those on page 3/4 3-28, are all related to modifications made to satisfy NUREG-0737 Item II.K.3.18 requirements and associated license condition 2.C.(28)(e).

In addition, the licensee has proposed in this amendment to correct errors contained in Table 3.3.3-1 Emergency Core Cooling System (ECCS) Actuation Logic Instrumentation (page 3/4 3-28) of the Technical Specifications. Footnote (a), where originally located on this page, applied to every entry. Footnote (a) states:

A channel may be placed in an operable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

This footnote was erroneously applied to the manual initiation functions, which can be performed without placing the required system in an inoperable status, and the Level 8 high pressure coolant injection trip function which would become unavailable if one channel were placed in an inoperable status. The proposed amendment would remove this footnote from these functions since it was incorrectly applied originally.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). One of these, Example (ii), involving no significant hazards considerations is "A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications, for example, a more stringent surveillance requirement." The requested changes to satisfy NUREG-0737, Item II.K.3.18 matches this example and the staff, therefore, proposes to determine that this change involves no significant hazards consideration.

Another of these, Example (i), involving no significant hazards consideration is "A purely administrative change to the Technical Specifications: For example a change to achieve consistency throughout the Technical Specifications, a correction of an error, or a change in nomenclature. The requested changes to correct the error related to the footnote in page 3/4 3-28 of Table 3.3.3-1 of the Technical Specifications clearly matches this example and the staff, therefore, proposes to determine that this change 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 & Trowbridge, 1800 M Street NW., Washington, D.C. 20036.

NRC Branch Chief: A Schwencer.

Pennsylvania Power & Light Company, Docket No. 50-387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of amendment request: January 31, 1985 as supplemented February 20, 1985.

Description of amendment request: In the proposed amendment the licensee has requested that: (1) Technical Specification 4/3 6.6.3 be revised to reflect the replacement of a one unit cooler subsystem with 2 recirculation fans to support drywell cooling improvements. The subject unit cooler subsystem will now be serving the general drywell area and the new recirculation fans will be supporting the safety-related function of post-LOCA drywell air mixing governed by this Technical Specification, (2) Technical Specification 4.8.4.1.a.1 be modified to achieve a greater level of clarity for this surveillance, which was previously ambiguous in cases where no trip setpoint or response time was provided. The difference between the current Technical Specification and the proposed revision is in specifying how acceptance criteria shall be met for each type of breaker, i.e., magneticonly (HFB-M) and thermal-magnetic (HFB-TM, KB-TM). The degree of testing for a given breaker remains unchanged due to the proposed revision. (3) Technical Specification Table 3.8.4.1-1 be revised to reflect the replacement of magnetic-only circuit breakers with thermal-magnetic circuit breakers. Changing the containment penetration over-current protection from magnetic-only to thermal-magnetic circuit breakers allows detection of substantially lower short circuit currents.

(4) Additional changes to Table 3.8.4.1-1 are deletion of the following:

Frame Rating /UL: Control of breaker frame size and UL rating is ensurd by the design change control process, which is governed by 10 CFR 50.59, and therefore the information need not be listed in the Technical Specifications.

Trip Setpoint: Due to the replacement of magnetic with thermal-magnetic circuit breakers, the number of adjustable (Type HFB-M, magnetic-only) breakers has decreased by approximately two-thirds. Trip setpoints are not applicable to non-adjustable breakers. The setpoint control of the adjustable breakers is ensured by the setpoint change control process, which is governed by 10 CFR 50.59, and therefore the information need not be included in the Technical Specifications.

Response Time: The response time column is currently "NA" for all entries. This is because, as described in FSAR subsection 4.8.4.1.a.1, manufacturer's data is used to determine acceptable response time. Therefore, the column has been deleted.

(5) Other changes to Table 3.8.4.1-1 are the following editorial changes:

a. "Circuit Breaker Location" has been changed to "Circuit Breaker Designation".

b. "Molded Case Circuit Breaker" headings were deleted. The need for this heading is tied, by the Standard Technical Specifications, to a need to differentiate test methods from those used for metal case circuit breakers. The surveillance is now tied to the types listed in the proposed change, and no metal case breakers are in use, so the deleted information serves no purpose.

c. Editorial description of specific equipment have been deleted. Systems and equipment numbers is sufficient for this purpose.

d. Footnotes referring to vendors have been deleted since they are unnecessary; the type definitions they provided are covered by the revised surveillance.

e. Footnote "+" was revised (new footnote*) to drop a reference to A and B, because this is not always the correct designation. Furthermore, such specific information is unnecessary; the key information is that two redundant breakers are to be OPERABLE.

f. For Type KB-TM, an informative footnote has been added since the breaker arrangement is atypical from the other types.

g. The entire listing has been reorganized to be grouped by system rather than randomly.

(6) *Drywell Cooling:* Two pairs of Type HFB-TM circuit breakers associated with drywell cooling have been added to the table to support recirculation fans 1V418A and B which are being added as discussed above.

Basis for proposed no significant hazards consideration determination: 1. In Technical Specification 3/4.6.6.3, these changes proposed support design improvements to the Drywell Atmosphere Recirculation and Cooling System. The only safety-related aspect of this change is that the post-LOCA air mixing, and the air flow capability of the new recirculation fans is the same as that of the unit cooler fans formerly used for this purpose. Furthermore, the equipment change is in accordance with existing design criteria and will not adversely affect the function of any system. Electric separation, seismic integrity and all other required design criteria are met. As discussed above, the safety function of post-LOCA mixing will be maintained. Furthermore, drywell cooling requirements specified in the Technical Specifications will be easier to maintain. Based on the above, the proposed changes do not: (1) Involve a significant increase in the probability or consequence of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in the margin of safety.

2. In Technical Specification 4.8.4.1.a.1 the degree of testing has not changed for any given breaker but, the amount of prescriptiveness required to clarify the acceptance criteria applicable to any given breaker has been increased. Therefore, this change falls under example (ii) "a change that constitutes an additional limitation, restriction or control not presently included in the Technical Specifications" of the Commission's guidance in (48 FR 14870) on types of amendments which are not likely to involve significant hazards considerations.

3. In Technical Specification Table 3.8.4.1-1 by replacing magnetic-only circuit breakers with thermal-magnetic circuit breakers safety has been improved by the addition of this equipment, which can detect lower short circuit currents. This design improvement provides additional control over penetration protection and it therefore falls under example (ii) "a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specification" of the Commission's guidance in (48 FR 14870) on types of amendments that are not likely to involve significant hazards considerations.

4. In Technical Specification Table 3.8.4.1-1 the licensee has proposed deletion of design information related to the overcurrent protection devices (Frame Rating/UL, Trip Setpoint and Response Time). This equipment is covered under the requirements of 10 CFR 50.59 which states that the licensee may make equipment changes without prior Commission approval, unless the proposed change involves a change to the Technical Specifications or an unreviewed safety question. Although the change would delete design information from the Technical Specifications this equipment would still be covered under 10 CFR 50.59 which does not permit any changes or replacement of equipment that is not the same as or equivalent to that currently installed. Therefore this deletion merely provides flexibility within the bounds of 10 CFR 50.59 without requiring changes to the Technical Specifications. The proposed changes therefore do not: (1) Involve a significant increase in the probability or consequence of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated or (3) involve a significant reduction in the margin of safety.

5. In Technical Specification Table 3.8.4.1-1 other changes have been proposed which are purely administrative and clearly fall under example (i) of actions not likely to involve significant hazards considerations, "A purely administrative change to Technical Specifications: For example, a change to achieve consistency throughout the Technical Specifications, correction of an error or a change in nomenclature.

6. In Table 3.8.4.1-1 additional circuit breakers were added to meet design criteria specified for the modifications described in Technical Specification 3/4 6.6.3 pertaining to drywell cooling to provide overcurrent protection for primary containment penetration conductors. Therefore the same basis for no significant hazards consideration as given for Technical Specification 3/4 6.6.3 (Item 1 above) is applicable.

On the basis of the above, the Commission proposes to conclude that all the proposed changes involve 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 & Trowbridge, 1800 M Street NW., Washington, D.C. 20036.

NRC Branch Chief: A. Schwencer.

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

Date of amendment request: January 31, 1985.

Description of amendment request: NUREG-0737 Item II.K.3.18 required modification of the automatic depressurization system (ADS) actuation logic to eliminate the need for manual actuation to assure adequate core cooling. The addition of the ADS manual inhibit switch to the Susquehanna Steam Electric Station (SSES), Unit 2, completes the modifications to the ADS needed to satisfy this requirement. Since the ADS is considered one of the safety systems used to assure emergency core cooling, license condition 2.C.(12)(f) required that the licensee propose Technical Specifications to cover the equipment installed that eliminated the need for manual actuation of the ADS. The changes to the Technical Specifications requested in this amendment, except those on page 3/4 3-28, are all related to modifications made to satisfy NUREG-0737 Item II.K.3.18 requirements and associated license condition 2.C.(12)(f).

In addition, the licensee has proposed in this amendment to correct errors contained in Table 3.3.3-1 Emergency Core Cooling System (ECCS) Actuation Logic Instrumentation (page 3/4 3-28) of the Technical Specifications. Footnote (a), where originally located on this page, applied to every entry. Footnote (a) states:

> A channel may be placed in an operable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

This footnote was erroneously applied to the manual initiation functions, which can be performed without placing the required system in an inoperable status, and the Level 8 high pressure coolant injection trip function which would become unavailable if one channel were placed in an inoperable status. The proposed amendment would remove this footnote from these functions since it was incorrectly applied originally.

Basic for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). One of these, Example (ii), involving no significant hazards considerations is "A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications, for example, a more stringent surveillance requirement." The requested changes to satisfy NUREG-0737, Item II.K.3.18 matches this example and the staff, therefore, proposes to determine that this change involves no significant hazards consideration.

Another of these, Example (i), involving no significant hazards consideration is "A purely administrative change to the Technical Specifications: For example a change to achieve consistency throughout the Technical Specifications, a correction of an error, or a change in nomenclature. The requested changes to correct the error related to the footnote in page 3/4 3-28 of Table 3.3.3-1 of the Technical Specifications clearly matches this example and the staff, therefore, proposes to determine that this change 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 & Trowbridge, 1800 M Street NW., Washington, D.C. 20036.

NRC Branch Chief: A Schwencer.

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

Date of amendment request: February 7, 1985.

Description of amendment request: The proposed amendment to the Technical Specifications (TSs) would revise the trip setpoint for isolation of the Reactor Core Isolation Cooling (RCIC) system on high steam line differential pressure. The current value for this trip setpoint was initially based on engineering judgment and operating experience. The proposed revised trip setpoint value is based on actual test data obtained using the startup test program.

Basis for proposed no significant hazards consideration determination: The intent of the trip setpoint in question is to insure isolation of the RCIC system occurs in the event of a design basis pipe break flow between 2.72 and 3.0 times maximum normal flow. Since the current pressure differential trip setpoint valve was based on engineering judgment and operating experience, it des not necessarily provide RCIC isolation within the design basis pipe break flow values. The proposed change would replace the current trip setpoint values based on actual inplant test data obtained during the startup test program. This would assure that RCIC isolation occurs within the desired design basis pipe break flow values. Since the proposed trip setpoints are based on actual test data and the current values are based on engineering judgment, no previous evaluations are compromised and no new accidents are created. In addition, the proposed values are more conservative than the current values, which increases the safety margin. On the basis of the above, the staff has determined that 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 previously evaluated; or (3) involve a significant reduction in a margin or safety. The staff therefore has made a proposed determination that this application for amendment involves no significant hazards consideration.

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

Attorney for licensee: Jay Silberg, Esquire, Shaw, Pittman, Potts & Trowbridge, 1800 M Street NW., Washington, D.C. 20036.

NRC Branch Chief: A. Schwencer.

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

Date of amendment request: October 1, 1984.

Description of amendment request: The proposed amendment would incorporate controls in the form of limiting condition for operation (LCO) into the Technical Specifications on equipment needed to insure proper functioning of the isolated 480 volt swing busses.

The licensee has identified a potential unreviewed safety question if appropriate controls are not placed on the power sources supporting the isloated 480 volt swing busses. In the current Technical Specifications for Susquehanna Steam Electric Station (SSES) Units 1 and 2 if any of these power sources are inoperative (i.e. down for maintenance) no LCO exists to require the swing busses to be returned to service within a specified time. A break in one recirculation line between the reactor vessel and the low pressure coolant injection loop (LPCI) in combination with a single failure of the 4kV power supply and an inoperative swing bus which both provide electrial power to the other loop results in a condition which renders both LPCI loops inoperable. Subsection 6.3.1.1.2 of the FSAR identifies the minimum combinations of emergency core cooling systems (ECCS) needed to recover from a pipe break in the primary system. Each combination requires at least one LPCI system to be operable. Since the scenario described does not satisfy the minimum ECCS requirement it represents a potential unreviewed safety question if the appropriate controls are not placed in the power sources supporting the isolated 480 volt swing bus. The proposed amendment will incorporate LCO's on the power sources supporting the isolated swing busses to prevent the potential for this scenario from occurring. The NRC staff has reviewed the Technical Specifications proposed by the licensee in this amendment and determined they are acceptable.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). One of these, Example (ii), involving no significant hazards considerations is "A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications: For example, a more stringent surveillance requirement." The requested change matches the example and the staff, therefore, proposes to characterize it as involving no signficant hazards consideration.

Local Public Document Room Location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 13701.

Attorney for licensee: Jay Silberg, Esquire, Shaw, Pittman, Potts & Trowbridge, 1800 M Street NW., Washington, D.C. 20036.

NRC Branch Chief: A. Schwencer.

Pennsylvania Power & Light Company, Docket No. 50-387 & 388, Susquehanna Steam Electric Station, Units 1&2, Luzerne County, Pennsylvania

Date of amendment request: February 11, 1985.

Description of amendment request: The proposed change to the Susquehanna Steam Electric Station (SSES) Unit 1 and Unit 2 Technical Specifications would permit the number of individuals on the Susquehanna Review Committee (SRC) to vary between eight and twelve and require a quorum, which consists of a majority of all members or designated alternatives approved by the Senior Vice President Nuclear, to be present for all formal meetings.

The Technical Specifications currently restrict the number of individuals on the SRC to nine. This "fixed number" restriction causes two problems: (1) When additional expertise is required, either a current voting member must be "replaced" temporarily or the more expert individual must be relegated to a non-voting status; and (2) when a vacancy is created on the current SRC roster, a replacement must immediately be found. This proposed change would provide additional flexibility, thereby relieving the above problems. Each new member chosen will still have to meet the qualification requirements stated in Technical Specification 6.5.2.2.

Basis for proposed no significant hazards consideration determination: The proposed changes are administrative improvements intended to provide additional flexibility related to the number of individuals that comprise the SRC. Their responsibilities are not diminished and the changes will not physically affect any safety related systems. The staff, therefore, proposes to conclude that the amendment to the Technical Specifications would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of an accident of a type different from any previously evaluated, or (3) involve a significant reduction in a margin of safety. On this basis, the staff has made an initial determination that the proposed amendment is not likely to involve a 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 & Trowbridge, 1800 M Street NW., Washington, D.C. 20036.

NRC Branch Chief: A. Schwencer.

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 13, 1981, as supplemented May 3, 1984, July 27, 1984 and January 18, 1985.

Description of amendment request: These submittals supplement the request for amendment dated July 13, 1981 which was noticed in the **Federal Register** on February 24, 1984 (49 FR 7040). The proposed revisions to the Technical Specifications would revise the testing requirements for hydraulic shock suppressors (snubbers) and add requirements for mechanical snubber operability and testing. The proposed changes were made in response to an NRC request to upgrade the testing requirements for all safety-related snubbers to ensure a higher degree of operability. The changes involve: Clarifying the frequency for visual inspection, stating the requirements for functional testing of snubbers which visually appear inoperable, the inclusion of a formula for the selection of representative sample sizes, the clarifying of the testing acceptance criteria, and revising the method of snubber listing to incorporate more information.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of these standards by providing certain examples (48 FR 14870). The examples of actions involving no significant hazards considerations include changes that constitute additional limitations or restrictions in the Technical Specifications. The proposed changes revise sections of the Technical Specifications related to hydraulic snubbers to clarify requirements and include additional testing, and incorporate both operability and testing requirements for mechanical snubbers. Since the requested changes upgrade the requirements for hydraulic snubbers and add requirements for mechanical snubbers, the staff proposes to determine that the application does not involve a significant hazards consideration.

Local Public Document Room Location: Penfield Library, State University College of Oswego, Oswego, New York.

Attorney for licensee: Mr. Charles M. Pratt, Assistant General Counsel, Power Authority of the State of New York, 10 Columbus Circle, New York, New York 10019.

NRC Branch Chief: Domenic B. Vassallo.

Power Authority of the State of New York, Docket No. 50-383, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: December 21, 1984, as supplemented February 19, 1985.

Description of amendment request: The proposed amendment would change the Technical Specifications (TS) to incorporate Radiological Effluent Technical Specifications (RETS) that would bring the license into compliance with Appendix I of 10 CFR Part 50. The proposed amendment would provide new Technical Specification sections pertaining to the following: Limiting conditions for operation and surveillance requirements for radioactive liquid, gaseous and solid wastes; total dose; and radiological environmental monitoring consisting of a monitoring program, land use census, and an interlaboratory comparison program. The proposed amendment would also incorporate into the Technical Specifications the bases that support the operation and surveillance requirements. In addition, some changes would be made in administrative controls, specifically delaing with the process control program, the offsite dose calculation manual, and the radiological monitoring program.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). One of the examples (ii) of actions not likely to involve a significant hazards consideration relates to changes that constitute additional restrictions or controls not presently included in the Technical Specifications.

The Commission, in a revision to Appendix I, 10 CFR Part 50, required licensees to improve and modify their radiological effluent systems in a manner that would keep releases of radioactive material to unrestricted areas during normal operation as low as is reasonably achievable. In complying with this requirement, it became necessary to add additional restrictions and controls to the Technical Specifications to assure compliance. This caused the addition of Technical Specifications described above. The staff proposes to determine that the application does not involve a significant hazards consideration since the changes constitute additional restrictions and controls that are not currently included in the Technical Specifications.

Location Public Document Room location: Penfield Library, State University College of Oswego, Oswego, New York.

Attorney for licensee: Mr. Charles M. Pratt, Assistant General Counsel, Power Authority of the State of New York 10, Columbus Circle, New York, New York 10019.

NRC Branch Chief: Domenic B. Vassallo.

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: January 30, 1985.

Description of amendment request: The proposed amendment would revise Table 3.7-1 "Process Pipeline Penetrating Primary Containment" on page 198 of Appendix A of the Technical Specifications (TS). The isolation signals for two reactor water sample line valves (drywell penetration X-41) would be changed from "B, C, D, E, and P" to "B and C".

The purpose of this change is to correct an error in Table 3.7-1 that was inadevertently introduced during the initial issuance of the TS. Three additional isolation signals "D, E, and P" were incorrectly included under the reactor water sample line entry in Table 3.7.1. This error was discovered during normal operation when the reactor water sample line isolation valves did not isolate on all signals listed in Table 3.7.-1.

Basis for proposed no significant hazards consideration determination: Signals B, C, D, E, and P effect closure of various Group A isolation valves. The Group A valves are located in process lines that communicate directly with the reactor vessel and penetrate the primary containment. Group A isolation functions include generation of isolation signals for the following components:

1. Main stream isolation valves (Penetration X-7A,B,C,D).
2. Main stream line drain isolation valves (Penetration X-8).
3. Reactor water sample line isolation valves (Penetration X-41).
4. Condenser vacuum pump.

The original plant design basis called for the reactor water sample line to isolate on signals B and C only. Signals, D, E, and P were intended to effect closure of the main stream isolation valves and main stream line drain isolation valves only. Therefore, the inclusion of the additional signals D, E, and P in Table 3.7-1 under drywell penetration X-41 represents an error.

The Commission has made a proposed determination that the 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. Clearly, the proposed amendment does not change the design basis of the plant but, rather, corrects an error to make the TS conform to the design basis. Therefore, the proposed amendment satisfies the three above-stated criteria for no significant hazards consideration.

Based on the foregoing, the Commission proposed to determine that the proposed amendment does not involve a significant hazards consideration.

Local Public Document Room location: Penfield Library, State University College of Oswego, Oswego, New York.

Attorney for licensee: Mr. Charels M. Pratt, Assistant General Counsel, Power Authority of the State of New York, 10 Columbus Circle, New York, New York 10019.

NRC Branch Chief: Domenic B. Vassallo.

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

Date of amendment request: May 3, 1983.

Description of amendment request: The amendment would revise and update Table 3.6-1 and Table 4.4-1 of the Technical Specification to reflect: (1) The replacement of double disc containment isolation value 850A with two single disc containment isolation valves 850A and 850C, (2) the installation of check valve 8406 as an automatic containment isolation valve, (3) the automation of valves 550, 863, 958, 959, and 1610, (4) the addition of two automatic containment valves, DW-AOV-1 and DW-AOV-2, for the demineralized water system. These modifications were made in response to position II.E.4.2.3 of NUREG-0737 which states that all non-essential systems be automatically isolated by the containment isolation signal.

The organizational changes requested in the May 3, 1983 submittal are being handled as a separate action and has received a separate Federal Register Notice (48 FR 52823).

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of these standards by providing examples (48 FR 14870). One of the examples of actions not likely to involve a significant hazards consideration relates to changes that constitute additional limitations or restrictions in the Technical Specifications. The proposed changes revise Table 3.6-1, the list of non-automatic containment isolation valves open continuously or intermittently for plant operation, and Table 4.4-1, the list of all containment isolation valves, to reflect modifications made in response to position II.E.4.2.3 of NUREG-0737. Since the requested changes upgrade the Technical Specifications to reflect stricter requirements for containment isolation valves, the staff proposes to determine that this application does not involve a 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 Branch Chief: Steven A. Varga.

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

Date of amendment request: December 3, 1984.

Description of amendment request: This amendment would revise Section 3.7 of the Technical Specifications to define the Limiting Conditions for Operation of systems, subsystems, trains, components and devices supplied by an inoperable normal or emergency power source, as provided by the Standard Technical Specifications.

By letter dated April 10, 1980, the staff issued a generic letter clarifying the term OPERABLE and requesting the licensee to propose Technical Specifications consistent with Model Technical Specifications. By Operating License Amendment No. 32, dated September 5, 1980, we granted the licensee's amendment, requested by letter dated May 23, 1980, implementing the Model Technical Specification definition of OPERABLE.

During a recent review it became clear that Amendment No. 32 did not fully satisfy the intent of the April 10, 1980 generic letter. By letter dated October 22, 1984, the staff requested that the licensee submit a license amendment to resolve the remaining concerns. This amendment request, dated December 3, 1984, is the licensee's response.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of these standards by providing examples (48 FR 14870). One of the examples (example (ii)) of actions not likely to involve a significant hazards consideration relates to changes that constitute an additional limitation, restriction, or control not presently included in the Technical Specifications. The proposed change, which revises Section 3.7 to define the Limiting Conditions for Operation of systems, subsystems, trains, components and devices supplied by an inoperable normal or emergency power source, falls into this category of additional limitations. Therefore, the staff proposes to determine that the application does not involve a significant hazards consideration.

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

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

NRC Branch Chief: Steven A. Varga.

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

Date of amendment request: January 17, 1985.

Description of amendment request: This amendment would revise the Technical Specifications related to steam generator tube inservice surveillance (Appendix A, Section 4.9 of the operating license) to extend the region for which the tube plugging limit of 63% degradation due to pitting applies. By letter dated November 9, 1984, the staff issued Amendment No. 50 granting an interim 63% plugging limit for the region from the tubesheet to the first subport plate for cold leg pitted tubes. The pending request would extend the region, for which the 63% degradation due to pitting limit applies, from the tubesheet to the second support plate for the remainder of Cycle 4.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of these standards by providing certain examples (48 FR 14870). An example (example (vi)) of an action likely to involve no significant hazards considerations is a change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan (SRP 5.4.2.2).

The licensee indicates that a detailed analysis, performed by Westinghouse, of similar units shows that extending the cold leg tube plugging limit of 63% to the regions above the first tube support plate and up to, but not including, the sixth support plate does not significantly change the accident analysis. Westinghouse has confirmed that this result conservatively applies to Indian Point 3. In addition, as stated in the Safety Evaluation for Amendment No. 50, previous plugging limits for cold leg pitted tubes were determined by the previous corrosion rate so as to maintain the requisite minimum wall thickness. Investigations made during the cycle 4 steam generator inspection outage indicate that the corrosion rate has significantly decreased. Therefore, the higher plugging limit would maintain the same minimum wall thickness.

The staff considers that this evidence meets the requirements of example (vi) and proposes to determine that the application does not involve a 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 Branch Chief: Steven A. Varga.

Portland General Electric Company, et al., Docket No. 50-344, Trojan Nuclear Plant, Columbia County, Oregon

Date of amendment request: January 29, 1985.

Description of amendment request: The amendment request was submitted in response to NRC Generic Letter 84-15 (dated July 2, 1984) which identified cold fast starts of diesel-generator sets as contributing to premature diesel engine degradation. In addition, excessive diesel engine testing was also identified as contributing to unnecessary wear. Consistent with the NRC request, the amendment would reduce the frequency of diesel-generator testing and allow the engine to be warmed up for most tests before increasing speed. The test starts from *ambient* conditions would be conducted semi-annually instead of monthly, consistent with the NRC guidance. Monthly testing would be continued.

The restriction that the 18-month testing be conducted during plant shutdown would be removed since some of the testing can be conducted while operating.

Finally, the time to shut down the plant in the event that two off-site transmission lines are not available would be changed from four hours to six hours. This is consistent with NRC guidance contained in NRC Standard Technical Specifications and consistent with other sections of the technical specifications that require a plant shutdown.

Basis for proposed no significant hazards consideration determination: The NRC staff has determined that excessive diesel-engine testing and cold fast starts contribute to premature engine degradation and that an overall improvement in reliability and availability can be gained by performing diesel-generator starts for surveillance testing using engine prelube and other manufacturer-recommended procedures to reduce engine stress and wear. The proposed amendment is consistent with this objective and therefore should result in enhanced reliability.

The change allowing six hours to shut down is minor. It is consistent with NRC guidance and would allow more time to conduct an orderly shutdown. Therefore it appears 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, or (3) involve a significant reduction in a margin of safety. Based on the foregoing, the NRC staff proposes to determine that the proposed amendment does not involve a significant hazards consideration.

Local Public Document Room location: Multnomah County Library, 801 SW. 10th Avenue, Portland, Oregon.

Attorney for licensee: J.W. Durham, Senior Vice President, Portland General Electric Company, 121 SW. Salmon Street, Portland, Oregon 97204.

NRC Branch Chief: James R. Miller.

Public Service Electric and Gas Company, Docket No. 50-311, Salem Nuclear Generating Station, Unit No. 2, Salem County, New Jersey

Date of amendment request: October 15, 1984.

Description of amendment request: The analysis of the reactor vessel material contained in surveillance capsule T, the first capsule to be removed for Salem Unit 2, showed that the transition temperature for the plate and weld material shifted more than predicted. Since the shifts were greater than predicted and the intermediate and lower shell vertical weld seam chemistries were estimated, the revised limits curves proposed in this amendment request are based on the upper limits of the Regulatory Guide 1.99 prediction curves. Specifically, the proposed amendment would: (1) Replace the present Heatup Limits Curve, Figure 3.4-2 with a new Heatup Limits Curve, (2) replace the present Cooldown Limits Curve Figue 3.4-3 with a new Cooldown curve, and (3) replace the present neutron fluence vs. Full Power Service Life, Figure B 3/4.4-1, with a new curve.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards for a No Significant Hazards determination by providing examples of actions not likely to involve a Significant Hazards Consideration in the **Federal Register** (48 FR 14870). One of the examples (ii) relates to changes that constitute additional limitations, restrictions, or controls not presently included in the technical specifications. Use of the proposed new curves, since they place more stringent limits on operation, will result in lower stresses to the Reactor Vessel during heatups and cooldowns.

Based on the above, since the proposed changes involve actions that conform to the referenced example in 48 FR 14870, we propose to determine that this application for amendment involves no significant hazards consideration.

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

Attorney for licensee: Conner and Wetterhann, Suite 1050, 1747 Pennsylvania Avenue, NW., Washingtion, D.C. 20006.

NRC Branch Chief: Steven A. Varga.

Public Service Electric and Gas Company, Docket No. 50-272, Salem Nuclear Generating Station, Unit No. 1, Salem County, New Jersey

Date of amendment request: October 15, 1984.

Description of amendment request: The proposed amendment would provide revised heatup and cooldown curves developed from the Capsule T analysis. Specifically, Technical Specifications Heatup and Cooldown Curves, Figures 3.4.3 and 3.4.3 would be replaced with revised figures.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards for a No Significant Hazards determination by providing examples of actions not likely to involve a Significant Hazard Consideration in the **Federal Register** (48 FR 14870). One of the examples (ii) related to changes that constitute additional limitations, restrictions, or controls not presently included in the technical specifications. Use of the proposed new curves, since they place more stringent limits on operations, will result in lower stresses to the Reactor Vessel during heatups and cooldowns.

Based on the above, since the proposed changes involve actions that conform to the referenced example in 48 FR 14870, we propose to determine that this application for amendment involves no significant hazards consideration.

Local Public Document Room location: Salem Free Library, 122 West Broadway, Salem, New Jersey 08079.

Attorney for licensee: Conner and Wetterhann, Suite 1050, 1747 Pennsylvania Avenue, NW., Washington, D.C. 20006.

NRC Branch Chief: Steven A. Varga.

Public Service Electric and Gas Company, Docket No. 50-311, Salem Nuclear Generating Station, Unit No. 2, Salem County, New Jersey

Date of amendment request: January 11, 1985.

Description of amendment request: The cycle 3 reload design for Salem Unit 2 is based on a revised control rod pattern which closely approximates the current Unit 1 control rod pattern. The design change request to reidentify control banks is being implemented during cycle 2-3 refueling outage, under the provisions of 10 CFR 50.59. The benefits associated with the revised rod pattern are as follows:

1. Reduces the maximum hot channel enthalpy rises factors during reactor maneuvers.

2. Provides a significant increase in operational flexibility by allowing an increase of the rod insertion limits.

Therefore, to take advantage of these benefits, this amendment request would change the Power Dependent Insertion Limit (PDIL) to allow a relaxation of the Unit 2 rod insertion requirements to match that of the current Unit 1 limits.

Basis for proposed no significant hazards consideration determination: A Reload Safety Evaluation (RSE) for cycle 3 hass been performed by Westinghouse and reviewed by Public Service Electric and Gas (PSE&G) to determine the impact of less restrictive (i.e. deeper) rod insertion limits. The results of the Westinghouse RSE show that the proposed limits do not cause the previously acceptable safety limits for any incident to be exceeded. PSE&G has review this analysis and concurs with the Westinghouse conclusions. The PSE&G reviewed consisted of performing an independent reload safety evaluation for cycle 3 using in-house computer codes, and it resulted in the determination that the current safety analysis design bases continue to be met.

The Commission has provided guidance concerning the application of the standards for a No Significant Hazards determination by providing examples of actions not likely to involve a Significant Hazards Consideration in the Federal Register (48 FR 14870). One of the examples (vi) relates to a change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan (SRP). As stated above, since the input parameters to the accident analyses are no less conservative than previously used values in the FSAR, the margins to safety remain at least as conservative with respect to the limits given for all analyzed accidents in Chapter 15.0 of the SRP and for appropriate sections of Chapter 4.3 of the SRP.

Based on the above evaluation we have determined that the proposed change in PDIL corresponds to example (vi) of guidance provided by the Commission in Federal Register 48 FR 14870, and the staff proposes to determine that this application for amendment involves no Significant Hazards Consideration.

Local Public Document Room location: Salem Free Library, 122 West Broadway, Salem, New Jersey 08079.

Attorney for licensee: Conner and Wetterhann, Suite 1050, 1747 Pennsylvania Avenue, NW., Washington, D.C. 20006.

NRC Branch Chief: Steven A. Varga.

Public Service Electric and 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 requests: February 8, 1985.

Description of amendment requests: The proposed amendment request would revise Salem Unit Nos. 1 and 2 Technical Specifications to agree with the attached corrected pages from Amendments 59 and 28 for Units 1 and 2 respectively. The corrections are predominantly typographical errors; several are editorial or clarifying in nature; finally, some meterial added by recent amendments has been inadvertently replaced by out-dated wording in the two year old license change which initiated Amendments 59 and 28.

Basis for proposed no significant hazards consideration determination: The proposed changes are administrative, in that, they either achieve consistency in the Technical specifications, add clarifications, or correct errors in the recently issued Amendments 59 and 28.

The Commission has provided guidance concerning the application of the standards for a No Significant Hazards determination by providing examples of actions not likely to involve a Significant Hazards Consideration in the Federal Register (48 FR 14870). One of the example (i) relates to a purely administrative change, for example the correction of an error. Since this proposed change conforms to this example, the Commission proposes to determine that the application for amendment does not involve a significant hazards consideration.

Local Public Document Room location: Salem Free Library, 122 West Broadway, Salem, New Jersey 08079.

Attorney for licensee: Conner and Wetterhann, Suite 1050, 1747 Pennsylvania Avenue, NW., Washington, D.C. 20006.

NRC Branch Chief: Steven A. Varga.

Public Service Electric and 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 requests: February 8, 1985.

Description of amendment requests: The amendments would make the following revisions to Technical Specifications Sections 3.3 and 3.11 as found in previous Amendments 59 and 28 for Salem Units 1 and 2 respectively:

1. On Table 3.3-12, TABLE NOTATION 28 should be modified on Unit No. 1 to base sampling/analysis requirements on containment fan coil unit operability.

2. On Table 3.3-12, TABLE NOTATION 28 should be modified on Unit No. 2 to allow for local monitor readout capabilities when control room indication is inoperable, and base sampling/analysis requirements on containment fan coil unit operability.

3. On Table 3.3-12, (Item 2.b) Instrument R-37, CHEMICAL WASTE BASIN LINE DISCHARGE, for Unit 2, change ACTION number to ACTION 31 and in the TABLE NOTATION add new ACTION 31 which bases sampling/analysis frequency on primary-to-secondary leak determination.

4. Delete Specification 3/4.11.2.6, GAS STORAGE TANKS to eliminate an unnecessary Curie limit on the Waste Gas Decay Tanks.

Basis for proposed no significant hazards consideration determination: Compared to the specifications that they affect, each of the four items above may, in some way, slightly reduce a safety margin by virtue of either decreasing a sampling frequency or by deleting an existing (albeit unnecessary) specification limit. However, operation of the Salem facilities with the proposed changes in place would remain clearly within all acceptable criteria specified in Standard Review Plan Sections 9.3.2 and 11.5 with respect to the affected systems and components.

The Commission has provided guidance concerning the application of the standards for a No Significant Hazards determination by providing examples of actions not likely to involve a Significant Hazards Consideration in the Federal Register (48 FR 14870). One of the examples (vi) relates to a change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan (SRP). Based on the above discussion we have determined that the four proposed changes to the Technical Specifications corresponds to example (vi) of guidance provided by the Commission in Federal Register 49 FR 14980, and the staff proposes to determine that this application for amendment involves no Significant Hazards Consideration.

Local Public Document Room location: Salem Free Library, 122 West Broadway, Salem, New Jersey 08079.

Attorney for licensee: Conner and Wetterhann, Suite 1050, 1747 Pennsylvania Avenue, NW., Washington, D.C. 20006.

NRC Branch Chief: Steven A. Varga.

Sacramento Municipal Utility District, Docket No. 50-312, Rancho Seco Nuclear Generating Station, Sacramento County, California

Date of amendment request: December 28, 1984.

Description of amendment request: This submittal revises the request for amendment dated September 9, 1982, which was noticed in the Federal Register on November 22, 1983 (48 FR 52825). The submittal: (1) Deletes inspection requirements that were completed during the 1983 refueling outage, (2) revises the Technical Specification paragraph numbers, and (3) adds a table to designate the selected special interest peripheral tubes that are to be eddy current tested.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards for a no significant hazards determination by providing certain examples (48 FR 14870). One of the examples (ii) of actions not likely to involve a significant hazards consideration relates to changes that constitute additional restrictions or controls not presently included in the Technical Specifications.

The September 9, 1982, application proposed Once Through Steam Generator (OTSG) Auxiliary Feedwater Header surveillances that were to be performed during the Rancho Seco 1983 refueling outage. These inspections were completed during the 1983 refueling outage. Therefore, the December 28, 1984, revision deleted these requirements from the proposed Technical Specifications and did not change the proposed Technical Specification surveillances to be conducted subsequent to the 1983 refueling outage. The paragraph numbers were changed because the original numbers had already been used in a Technical Specification revision issued after September 9, 1982. The September 9, 1982, submittal specified that special interest peripheral tubes would be eddy current inspected. The December 28, 1984, submittal added a table to designate which tubes are the special interest peripheral tubes. Thus, the December 28, 1984, submittal revises the proposed Technical Specification change to: (1) Delete a surveillance already completed, (2) administatively change paragraph numbers, and (3) add an additional restriction not presently included in the Technical Specifications.

Therefore, since the application for amendments consists of an additional restriction not presently included in the Technical Specifications, the Commission's previous proposed determination, that the application for amendment does not involve a significant hazards consideration, remains unchanged.

Local Public Document Room location: Sacramento City-County Library, 828 I Street, Sacramento, California.

Attorney for licensee: David S. Kaplan, Sacramento Municipal Utility District, 6201 S Street; P.O. Box 15830, Sacramento, California 95813.

NRC Branch Chief: John F. Stolz.

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

Dates of amendment request: August 24 and November 14, 1984.

Description of amendment request: The amendment would change time constant T_4 in the overtemperature delta-T setpoint equation from 33 seconds to 28 seconds and would change the reactor trip setpoint for the steam generator water level low-low signal. Currently, this setpoint is linear from 12% to 54.9% of span for 30% to 100% of rated thermal power (RTP). This would be changed to 12% to 30% of span for 30% to 100% of RTP. Also, the allowable value associated with the trip setpoint is being changed a corresponding amount.

Basis for proposed no significant hazards consideration determination: Reducing the value of T_4 from 33 seconds to 28 seconds will slow down the response to the T-average dynamic compensation of the overtemperature delta-T setpoint. The dynamic T-average term in the overtemperature delta-T equation compensates for inherent instrument response times and piping transport lags between the core and the temperature sensors in the manifolds. This reduction in T_4 lowers the lead/lag ratio by 15% resulting in a comparable reduction in the anticipatory response of the T-average compensation of the setpoint.

The seven safety analyses correlated with the overtemperature delta-T trip have been reviewed for the effect of the new T_4 and found to still be acceptable. Four of the safety analyses are more conservative with this change. The effect of the decrease in T_4 on the three remaining analyses that take credit for the overtemperature delta-T trip is dicussed below for each transient.

Protection for the rod withdrawal at power accident is provided by the overtemperature delta-T trip for low reactivity insertion rates and by the high neutron flux trip for high reactivity insertion rates (FSAR Figure 15.2.8). The decrease in T_4 will cause the point at which the two segments of the curve in Figure 15.2.8 meet to be at a slightly lower reactivity insertion rate. The high neutron flux portion of the curve remains above the limiting departure from nucleate boiling ratio (DNBR) of 1.30.

Uncontrolled boron dilution events require operator action to recognize and terminate the uncontrolled dilution. For an uncontrolled boron dilution at power, the analysis assumes that the operator is alerted to the event by the overtemperature delta-T reactor trip. The analysis indicates that the operator has 43.2 minutes after the trip to terminate the dilution. The decrease in T_4 will result in an insignificant delay in receiving the overtemperature delta-T trip and therefore the response time will not be significantly decreased. The delay is small because the rate of increase in T-average is very slow for a boron dilution event resulting in very little dynamic compensation of the setpoint. The operator response time will still be approximately 43 minutes: more than ample time for the operator to recognize and terminate the event.

Protection for the loss of load accident is provided by the overtemperature delta-T trip when pressurizer pressure control is assumed to function and by the high pressurizer pressure trip when pressurizer pressure control is assumed not to function. FSAR Section 15.2.7 documents the results of analyses for each of these assumptions considering both beginning of life and end of life conditions. For the beginning of life case (small negative moderator temperature coefficient) with pressurizer pressure control, the decrease in T_4 results in a small delay in the overtemperature delta-T trip and a slightly lower minimum DNBR of approximately 1.50 which is still well above the acceptance criteria of 1.30 (FSAR Figure 15.2-19).

For the end of life case (large negative moderator temperature coefficient) with pressurizer pressure control, the decrease in T_4 again results in a small delay in the overtemperature delta-T trip, howevr, DNBR does not decrease below its initial value, but increases (see FSAR Figure 15.2-21). The increase in DNBR is due to the decrease in nuclear power from the negative moderator temperature coefficient and the increase in pressurizer pressure.

The above discussion demonstrate that the effect of the decrease in T_4 on the protection provided by the overtemperature delta-T reactor trip is minimal and that the safety analysis design basis will continue to be met.

The Commission has provided certain examples (48 FR 14870) of actions likely to involve no significant hazards considerations. One of the examples (vi) relates to a change which may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system specified in the Standard Review Plan.

The change to time constant T_4 is similar to this example. Accordingly, the Commission proposes to determine that this change does not involve significant hazards considerations.

The basis function of the reactor protection circuits associated with low steam generator water level is to preserve the steam generator heat sink for removal of long term residual heat. Therefore, the low-low steam generator water level trip is provided for each steam generator to ensure that sufficient initial thermal capacity is available in the steam generator at the start of the transient due to the loss of normal feedwater accident.

The loss of normal feedwater accident analysis was reviewed for the effect of the change in the low-low steam generator water level trip from 54.9% to 30% of span at 100% rated thermal power. It was found that the original accident analysis assumed a 0% of span trip setting. Therefore, even assuming worst case instrument uncertainties, the 30% of span trip setpoint provides 5.2 feet more thermal capacity than that assumed in the safety analysis.

The Commission has provided certain examples (48 FR 14870) of actions likely to involve no significant hazards considerations. The request involved in this case does not match any of those examples. However, the staff has reviewed the licensee's request for the above amendment and has determined that should this request be implemented, it will not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated because the lose of normal feedwater accident is not made more probable and the accident analysis design basis continues to be met, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated because the trip setpoint setting is still covered by the original accident analysis. Also, it will not (3) involve a significant reduction in a margin of safety because a steam generator heat sink greater than that assumed in the original accident analysis will continue to exist. Accordingly, the Commission proposes to determine that this change doe not involve significant hazards considerations.

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

Attorney for licensee: Randolph R. Mahan, South Carolina Electric and Gas Company, P.O. box 764, Columbia, South Carolina 29218.

NRC Branch Chief: Elinor G. Adensam.

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

Date of amendment request: February 14, 1985.

Decription of amendment request: This amendment would approve changes to the Technical Specifications regarding testing of the emergency diesel generators. The proposed changes would: (1) limit diesel engine loading to 4500 kW plus or minus 5% for engine testing and emergency service requirements, (2) eliminate fast engine starts from the monthly surveillance testing, but retain the refueling interval fast start test which simulates design basis emergency power requirements, (3) delete the requirement to run the diesel generators for 60 minutes at 4422 kW load during the refueling interval test (TS 4.4.F.2(d)), and (4) specify that the monthly surveillance and refueling interval tests start from "standby conditions" rather than "ambient conditions".

Basis for proposed no significant hazards consideration determination: The NRC staff's safety evaluation "Transamerica Delaval, Inc. (TDI) Diesel Engine Reliability and Operability—San Onofre Nuclear Generating Station, Unit 1," dated November 19, 1984, requested that the licensee, among other items, propose Technical Specification changes to accomplish the following: Engine load shall not exceed 4500 kW plus or minus 5% for engine testing and emergency service requirements, and monthly surveillance testing will not include "fast starts," but rather "slow starts." The refueling interval "fast start" test which simulates design basis service requirements should be retained.

Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," dated July 2, 1984, also discusses the reduction in number of cold fast start surveillance tests for diesel generators.

Items (1) and (2), discussed above, were proposed by the licensee to meet the Technical Specification changes requested by the staff.

The licensee also proposed two additional changes to the Technical Specifications (Items 3 and 4, above). TS 4.4.F.2(d) currently requires that the diesel generator run for 60 minutes at 4422 kW load as part of the test that simulates safety injection demand concurrent with loss of offsite power. The licensee has proposed to delete this Technical Specification and has stated that it is considered superfluous in view of TS 4.4.F.2(b) which verifies the capability of the diesel to automatically take on emergency loads and then run for 5 minutes. Technical Specifications 4.4.B.1 and 4.4.F.2 currently specify that the monthly surveillance and refueling interval diesel generator test start shall be from "ambient" conditions. The proposed change would specify that these starts shall be from "standby" conditions. The licensee stated that "ambient" is considered misleading for a diesel generator system that is normally maintained above ambient temperature.

The licensee has made a no significant hazards consideration determination pursuant to 10 CFR 50.92. The licensee stated that the proposed changes will ensure that loading of the diesel generators for monthly surveillance tests and refueling interval tests is realistic and not excessive, and unnecessary fast test starts are avoided. These changes will minimize mechanical stress and wear of the engine components and therefore prolong engine life. The licensee further stated that the proposed change will not significantly impact the effectiveness of surveillance testing and refueling interval testing, nor will it reduce the frequencies of any of these tests, nor impact the current availability of the diesel generators in all modes of plant operation. In light of these considerations, the licensee concluded that the proposed changes will enhance plant safety. Thus, the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated because the changes are designed to prolong the diesel engine life and will not affect the frequency of engine tests nor impact the availability of the diesel generators. The proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated because the changes will not impact the current availability of the diesel generators in all modes of plant operation. The proposed changes also do not involve a significant reduction in a margin of safety because the changes are designed to prolong engine life, will not significantly impact the effectiveness of testing, will not reduce the frequency of the test nor impact the availability of the diesel generators.

The staff has reviewed the licensee's significant hazards consideration determination, and based on this review, the staff has made a proposed determination that the application for amendment involves no significant hazards consideration.

Local Public Document Room location: San Clemente Public Library, 242 Avenida Del Mar, San Clemente, California 92672.

Attorney for licensee: Charles R. Kocher, Assistant General Counsel, James Beoletto, Esquire, Southern California Edison Company, Post Office Box 800, Rosemead, California 91770.

NRC Branch Chief: John A. Zwolinski.

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

Date of amendment request: December 13, 1984.

Description of amendment request: The amendment would modify paragraph 6.2.B.4 of the Appendix A Technical Specifications to:

(1) Delete a requirement that unreviewed safety questions be reviewed by the Plant Operations Review Committee (PORC). (Unreviewed safety questions determinations would still be reviewed by the Nuclear Safety Review Board.)

(2) Change the requirement that Radiological Emergency Plan procedures and Industrial Security Program procedures be reviewed annually by the PORC. (The licensee proposes that the PORC review each procedure once and subsequently review only changes to procedures.)

(3) Delete a requirement that the PORC review the adequacy of employee training programs and recommend changes. (Training programs will continue to be audited annually by the Nuclear Safety Review Board.)

Basis for propssed no significant hazards consideration determination: The Commission has provided guidance for the application of criteria for no significant hazards consideration determination by providing examples of amendments that are considered not likely to involve significant hazards considerations (48 FR 14870). These examples include: "(vi)— A change which either may result in some increase to the probability or consequences of a previous-analyed accident or reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan (SRP): For example, a change resulting from the application of a small refinement of a previously-used calculational model or design method."

The proposed changes will reduce the extend of management overview of certain safety-related activities and may thereby reduce a safety margin. However, the revised requirements would be consistent with ANSI/ANS.3.2, "American National Standard, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants" which is endorsed by SRP Chapter 17.2. The proposed changes are thus encompassed by example (vi) of the Commission's guidance.

Since the application for amendment involves proposed changes that are encompassed by the criteria or an example for which no significant hazards consideration exists, the staff has made a proposed determination that the application involves no significant hazards consideration.

Local Public Document Room location: Athens Public Library, South and Forrest, Athens, Alabama 35611.

Attorney for licensee: H. S. Sanger, Jr., Esquire, General Counsel, Tennessee Valley Authority, 400 Commerce Avenue, E 11B 33C, Knoxville, Tennessee 37902.

NRC Branch Chief: Domenic B. Vassallo.

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

Date of amendment request: February 25, 1985.

Description of amendment request: The amendments would delete paragraphs 3.5.B.11 through 3.5.B.13 and 4.5.B.11 through 4.5.B.13 of the Appendix A Technical Specifications. These paragraphs specify limiting conditions for operation and surveillance requirements associated with the residual heat removal (RHR) system crossties between adjacent reactor units. These crossties provide for certain RHR pumps and heat exchangers in each unit to serve as backups for those in an adjacent unit for long term shutdown cooling and permit fluid makeup from the adjacent unit.

By proper valve alignment, the network created by the RHR crossties permits the B (or D) RHR pumps on Unit 1 to circulate Unit 2 suppression pool or reactor vessel water through the B (or D) heat exchangers on Unit 1 in the event that the Unit 2 RHR pumps are unavailable. The crosstie network is sized for a minimum flow of 5,000 gpm which will achieve about 91% of full flow heat transfer capability of the RHR heat exchangers. In a like fashion, the A (or C) RHR pumps on Unit 2 can be used to circulate Unit 1 suppression pool or reactor vessel water through the A (or C) heat exchangers on Unit 2. The B (or D) RHR pumps on Unit 2 and the A (or C) RHR pumps on Unit 3 can be similarly utilized. Suppression pool water which has been circulated through the RHR heat exchangers on one unit can be used to flood the reactor core, spray the drywell and suppression chamber, or returned to the suppression chamber of the adjacent unit. In this way decay heat and residual heat can be removed from the reactor core and primary containment of the adjacent unit on a long term basis.

The operability requirements for crosstie RHR cooling capability require, during certain maintenance or modification activities in one unit, that other units be shutdown (i.e., if Unit 2 RHR systems are taken out of service, Units 1 and 3 must be shutdown within 10 days). Deletion of the Technical Specifications will permit adjacent units to continue operating during RHR system modification and maintenance activities.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance for the application of criteria for no significant hazards consideration determination by providing examples of amendments that are considered not likely to involve significant hazards considerations (48 FR 14870). These examples include: "(vi) A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan (SRP): For example, a change resulting from the application of a small refinement of a previously-used calculational model or design method."

The proposed changes may reduce the redundancy available to the RHR system and may thereby increase the probability or consequences of accidents which are mitigated by the RHR system. However, no credit was given for the RHR crosstie feature in the facilities' licensing basis (Final Safety Analysis Report (FSAR) Section 4.8), and, without the feature the RHR system will still meet the redundancy requirements of the acceptance criteria of SRP Section 5.4.7, "Residual Heat Removal System." The proposed changes are thus encompassed by example (vi) of the Commission's guidance.

Since the application for amendments involve proposed changes that are encompassed by an example for which no significant hazards consideration exists, the staff has made a proposed determination that the application involves no significant hazards consideration.

Local Public Document Room location: Athens Public Library, South and Forrest, Athens, Alabama 35611.

Attorney for licensee: H. S. Sanger, Jr., Esquire, General Counsel, Tennessee Valley Authority, 400 Commerce Avenue, E 11B 33C, Knoxville, Tennessee 37902.

NRC Branch Chief: Domenic B. Vassallo.

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

Date of amendment request: September 17, 1984.

Description of amendment request: The proposed amendment would add a requirement to the Administrative Controls Section of the Appendix A Technical Specifications to report, on a monthly basis, all challenges to the Pressurizer power operated relief valve (PORV) and Pressurizer code safety valves. The amendment request is submitted in response to a request from the NRC.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). The examples of actions involving no significant hazards consideration include actions which involve a change that constitutes an additional limitation, restriction or control not presently included in the Technical Specifications. The proposed change matches this example since the above reporting requirement is not presently included in the Technical Specifications. Therefore, the staff proposes to determine that the application does not involve a significant hazards consideration.

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

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts, and Trowbridge, 1800 M Street, NW., Washington, D.C. 20036.

NBC Branch Chief: John F. Stolz.

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

Date of amendment request: October 8, 1984.

Description of amendment request: The proposed amendment would add a new section (Section 6.2.3) to the facility Technical Specifications. The new section would add a requirement for administrative procedures to limit the working hours of facility staff who perform safety related functions. The procedures would limt the amount of overtime worked by the facility staff, such as senior reactor operators, reactor operators, auxiliary orperators, health physicists, and key maintenance personnel, in accordance with guidelines included with the new Technical Specification section. The proposed amendment request was submitted in response to NUREG-0737, Item I.A.1.3.1, and Generic Letter 82-12.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). The examples of actions involving no significant hazards consideration include actions which involve a change that constitutes an additional limitation, restriction or control not presently included in the Technical Specifications. The proposed change matches this example since limitations on working hours are not covered in the current Technical Specifications. This additional requirement would enhance safe plant operation by limiting overtime worked by key personnel so that the potential for human error caused by fatigue can be reduced. Therefore, the staff proposes to determine that the application does not involve a significant hazards consideration.

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

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts, and Trowbridge, 1800 M Street, NW., Washington, D.C. 20036.

NBC Branch Chief: John F. Stolz.

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

Date of amendment request: November 5, 1984.

Description of amendment request: The proposed amendment would change the Davis-Besse Appendix A Technical Specification Surveillance Requirement 4.7.1.2.d by deleting the requirement for the Auxiliary Feed Pump Turbine Inlet Steam Pressure Interlocks to be demonstrated operable. This change, in effect, would permit the deletion of these interlocks from the system.

Basis for proposed no significant hazards consideration determination: The pressure interlocks were installed in the plant to protect againt the effects of a rupture of the steam supply line to the auxiliary feed pump turbines. Under the guidance existing at the time, the line was classified as a high-energy line. Subsequently, a Standard Review Plan and Branch Technical Position were published which would permit application of moderate-energy criteria to high-energy lines which are used infrequently, not more than 2% of the plant operating time. Evaluation under the less stringent criteria would not require the pressure interlocks for protection and would permit their deletion. Elimination of these interlocks will improve reliability of the auxiliary feedwater system.

The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). An example of an amendment not likely to involve a significant hazards consideration is (example (vi)), a change which may result in some increase to the probability or consequences of a preiously-analyzed accident or reduce in some way a safety margin but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan.

The proposed change matches this example; therefore, the staff proposes to determine that the application does not involve a sigificant hazards consideration.

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

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts and Trowbridge, 1800 M Street, NW., Washington, D.C. 20036.

NRC Branch Chief: John F. Stolz.

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

Date of amendment request: December 18, 1984.

Description of amendment request: The proposed amendment would add the title of "Nuclear Training Manager" to Technical Specification Section 6.4.1. This Section currently does not indicate the position title of the individual responsible for the direction of the retraining and replacement training program for facility staff.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). The examples of actions not likely to involve a significant hazards consideration include actions related to a purely administrative change to the Technical Specifications such as a change to achieve consistency throughout the Technical Specifications, correcting an error, or a change in nomenclature. The proposed change matches this example since the addition of the position title only remedies the previous omission in the Technical Specifications and in no way affects the conduct or effectiveness of the training program itself. Therefore, the staff proposes to determine that the application does not involve a significant hazards consideration.

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

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts, and Trowbridge, 1800 M Street, NW., Washington, D.C. 20036.

NRC Branch Chief: John F. Stolz.

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

Date of amendment request: December 16, 1984.

Description of amendment request: The amendment would modify the Technical Specifications to incorporate revisions to reporting requirements in response to Generic Letter 83-43 to comply with 10 CFR 50.72 and 50.73. Changes are made in Definitions, Instrumentation, Reactor Coolant System, Plant Systems and Administrative Controls Sections of the Appendix A Technical Specifications.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards for making a no significant hazards consideration determination by providing certain examples (48 FR 14870). The example which the proposed amendment fits is: "(vii) A change to make a license conform to changes in the regulations, where the license change results in very minor changes to facility operations clearly in keeping with the regulations." The Commission revised 10 CFR 50.72 and added 10 CFR 50.73, both to become effective January 1, 1984. These regulations revised the immediate notification requirements for operating nuclear power reactors and revised the Licensee Event Report System. The Commission then provided to the licensee model Technical Specifications to incorporate these regulation changes. The licensee has now proposed the changes in Technical Specifications to comply with the regulations. For these reasons, the Commission proposes to determine that the amendment involves no significant hazards consideration.

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

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts, and Trowbridge, 1800 M Street, NW., Washington, D.C. 20036.

NRC Branch Chief: John F. Stolz.

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

Date of amendment request: January 10, 1985.

Description of amendment request: The purpose of the proposed amendment request is to revise Technical Specification Figures 6.2-1 and 6.2-2 to include modified organizational charts in the Administrative Technical Specifications. The proposed change reduces the detail in the Figures and results in more generic Technical Specification organizational charts, without reducing commitments and without conflict to the organization as described in Technical Specification Section 6.0. This request to reduce the amount of detail in the organizational charts does not represent a change in reporting relationships; a change in responsibilities; or a change in commitments. Positions deleted from the Technical Specification charts are still described in the Figures and text of Chapter 13 of the Final Safety Analysis Report.

Basis for proposed no significant hazards consideration determination: The licensee, by letter dated January 10, 1985, stated that the proposed change does not: (1) Involve a significant increase in the probability or consequences of an accident or other adverse condition over previous evaluations; or (2) create the possibility of a new or different kind of accident or condition over previous evaluations; or (3) involve a significant reduction in a margin of safety. The Commission has provided guidance concerning the application of the Standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). One of the examples of actions involving no significant hazards consideration relates to a purely administrative change to Technical Specifications. This amendment request reduces the detail in Technical Specification Figures 6.2-1 and 6.2-2 to produce more generic Technical Specification organizational figures, without reducing commitments and without conflict to the organizational description in Section 6.0 of the Tech. Specs. The revised charts do not represent a change in reporting relationships; a change in responsibilities; or a change in commitments. The organization remains essentially as previously submitted and approved. Based on the foregoing, the requested amendment does not present a significant hazard.

Local Public Document Room locations: Fulton City Library, 709 Market Street, Fulton, Missouri 65251 and the Olin Library of Washington University, Skinker and Lindell Boulevards, St. Louis, Missouri 63130.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 1800 M Street, NW., Washington, D.C. 20036.

NRC Branch Chief: B. J. Youngblood.

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

Date of amendment request: January 29, 1985.

Description of amendment request: The purpose of the proposed amendment request is to revise the surveillance requirements given in Technical Specification Table 4.3-1 in order to comply with the following staff requirements.

(a) Independent, on-line testing of the Undervoltage and Shunt Trip Attachments on the reactor trip breakers per Item 4.5.1 of Generic Letter 83-28; and

(b) Periodic testing of the Undervoltage and Shunt Trip Attachments on the bypass breakers, with test intervals as defined by the staff.

The addition of the above surveillance requirements results in greater confidence that the respective Undervoltage and Shunt Trip Attachments will perform as designed.

Basis for proposed no significant hazards consideration determination: The licensee, by letter dated January 29, 1985, stated that the proposed change does not: (1) Involve a significant increase in the probability or consequences of an accident or other adverse condition over previous evaluations; or (2) create the possibility of a new or different kind of accident or condition over previous evaluations; or (3) involve a significant reduction in a margin of safety. The Commission has provided guidance concerning the application of the Standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). This amendment request is similar to the example of an action involving no significant hazards consideration which relates to a change that constitutes an additional limitation, restriction or control not presently included in the Technical Specifications. This amendment request involves the addition of two surveillance requirements, as requested by the staff, which result in greater confidence that the respective Undervoltage and Shunt Trip Attachments will perform as designed. Based on the foregoing, the requested amendment does not present a significant hazard.

Local Public Document Room locations: Fulton City Library, 709 Market Street, Fulton, Missouri 65251 and Olin Library of Washington University, Skinker and Lindell Boulevard, St. Louis, Missouri 63130.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 1800 M Street NW., Washington, D.C. 20036.

NRC Branch Chief: B.J. Youngblood.

Washington Public Power Supply System, Docket No. 50-397, WNP-2, Richland, Washington

Dates of amendment request: December 20, 1984 and January 31, 1985.

Description of amendment request: Currently the WNP-2 Technical Specification requires at least one containment air lock door to be closed at all times during plant operation and locked closed within 24 hours if the other door becomes inoperable. The Specification is silent with respect to the interlock mechanism that assures that only one air lock door can be open at any one time.

The proposed amendment to Operating License NPF-21 would revise the WNP-2 Technical Specifications to permit repair and/or maintenance of the interlock mechanism for the primary containment and/or maintenance of the interlock mechanism for the primary containment air locks during plant operations. The amendment is intended to assure containment integrity in the event the containment air lock itself is operable—but the airlock door interlock mechanism is inoperable—by substituting administrative controls to ensure that at least one door is closed at all times while the mechanical interlock mechanism is undergoing repair or maintenance. The purpose of the proposed change is to amplify and clarify the Technical Specification in this regard.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility 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; (2) create the possibility of a new or different kind of accident from an accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Subsequent to his initial (December 20, 1984) request, the licensee has determined that the requested amendment per 10 CFR 50.92 does not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated because the administrative controls serve the same function as the mechanical interlock; namely, to ensure containment integrity is maintained; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated also because the administrative controls serve the same function as the mechanical interlock so there is no new or different kind of accident scenario; or

(3) Involve a significant reduction in a margin of safety because the requirement for leak tightness remains unchanged.

Based on staff review of the proposed changes, we find that there exists reasonable assurance that containment integrity will not be violated whenever the interlock mechanism or one air lock door becomes inoperable provided proposed administrative controls are instigated. The proposed administrative controls include:

1. Assignment of a dedicated individual to assure that both air lock doors cannot be opened simultaneously whenever the air lock is used and the interlock mechanism is inoperable; and

2. Locking closed one of the air lock doors that remains operable if the interlock mechanism or air lock door cannot be returned to service within 24 hours; and

3. Verifying that an uperable air lock door is locked closed prior to each closing of the shield door and at least once per shift while the shield door is open.

NB: Outside the containment, the air lock is completely enclosed in a shield cubicle that has a door (shield door) with two locks on it. Opening of the shield door is alarmed in both the control room and the central alarm station.

The licensee has determined and the NRC staff agrees that these changes have little safety significance and that the proposed amendment will not alter any of the accident analyses.

Accordingly, the Commission proposes to determine that the proposed changes to the Technical Specifications involve no significant hazards considerations.

Local Public Document Room location: Richland City Library, Swift and Northgate Streets, Richland, Washington.

Attorney for licensee: Nicholas Reynolds Esquire, Bishop, Cook, Liberman, Purcell and Reynolds, 1200 Seventeenth Street, NW., Washington, D.C. 20036.

NRC Branch Chief: A. Schwencer.

Wisconsin Electric Power Company, Docket No. 50-266, Point Beach Nuclear Plant, Unit No. 1, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendment request: January 11, 1985.

Description of amendment request: The amendment would modify the license to delete conditions imposed by the Commission's Confirmatory Order for Modification of License dated November 30, 1979 and Order Modifying Confirmatory Order dated January 3, 1980. The conditions of those Orders which are currently in effect are:

1. Primary coolant activity for Point Beach Nuclear Plant Unit 1 will be limited in accordance with the provisions of Sections 3.4.8 and 4.4.8 of the Standard Technical Specifications for Westinghouse Pressurized Water Reactors, Revision 2, July 1979, rather than Technical Specification 15.3.1.C.

2. Close surveillance of primary-to-secondary leakage will be continued and the reactor will be shut down for tube plugging on detection and confirmation of any of the following conditions:

(a) Sudden primary-to-secondary leakage of 150 gpd (0.1 gpm) in either steam generator.

(b) Any primary-to-secondary leakage in excess of 250 gpd (0.17 gpm) in either steam generator.

(c) An upward trend in primary-to-secondary leakage in excess of 15 gpd (0.01 gpm) per day when measured primary-to-secondary leakage is above 150 gpd.

3. The reactor will be shut down, any leaking steam generator tubes plugged, and an eddy current examination

performed if any of the following conditions are present:

(a) confirmation of primary-to-secondary leakage in either steam generator in excess of 500 gpd (0.35 gpm).

(b) Any two identified leaking tubes in any 20 calendar day period.

This eddy current program will be submitted to the NRC for staff review.

4. The NRC staff will be provided with a summary of the results of the eddy current examination performed under item 3 above, including a description of the quality assurance program covering tube examination and plugging. This summary will include a photograph of the tubesheet of each steam generator which will verify the location of tubes which have been plugged.

5. The licensee will not resume operation after the eddy current examinations required to be performed in accordance with condition 3 above until the Director, Office of Nuclear Reactor Regulation determines in writing that the results of such tests are acceptable.

6. Unit 1 will not be operated with more than 18% of tubes plugged in either steam generator.

7. Unit 1 will be operated at a reactor coolant pressure of 2000 psia with the associated parameters (i.e., overtemperature delta T and low pressurizer pressure trip point) with limits indicated in the Safety Evaluation Report appended to the Orders.

The above conditions (with the exception of items 1 and 7 which are currently included in the Technical Specifications) would be deleted by the proposed amendments and the requirements of Technical Specification 15.3.1.D would be in effect.

Basis for proposed no significant hazards consideration determination: The operating restrictions imposed by the Commission's Orders were necessary because of the severely degraded nature of Point Beach Unit 1 steam generators. Those steam generators were replaced in late 1983-early 1984 with new lower internals including tube bundles and refurbished upper internals. The staff's Safety Evaluation of July 15, 1983 concluded that replacement of the steam generators for Point Beach Unit 1 could be conducted safety and that continued operation with the replaced steam generators would not pose a threat to the public health and safety. The staff's environmental review was completed on September 30, 1983 and in it the staff concluded that replacement of the Unit 1 steam generators and subsequent operation would have neither significant radiological nor non-radiological environmental impact. The new steam generators have been operating satisfactorily for approximately one year. The staff has concluded that the restrictions imposed by the Commission's Orders are no longer necessary with the exception of those items incorporated into the Technical Specifications which will not be changed by the proposed amendment. Because the replaced steam generators are not significantly different than those previously in place at Point Beach Nuclear Plant except where the changes were reviewed and approved in the staff's Safety Evaluation the staff finds that the proposed amendments 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. Therefore, the staff proposes to determine that the amendment involves no significant hazards considerations.

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

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 1800 M Street NW., Washington, D.C. 20036.

NRC Branch Chief: James R. Miller.

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: January 30, 1985.

Description of amendment request: The proposed amendments would revise the Technical Specifications covering the low frequency trip setpoints for the reactor coolant pump motor breakers. Specifically, the low frequency trip setpoint would be changed from 57.5Hz to 55.0Hz.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards for determining whether a significant hazard exists by providing certain examples (48 FR 14870). One of the examples of actions involving no significant hazards considerations is example (vi). "a change which either may result in some increase to the probability or consequences of a previously analyzed accident, or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan".

The licensee has provided the results of a reanalysis of the loss of flow transient, performed by Westinghouse, in support of its proposed amendment application. The analysis methodology and assumption for this transient are consistent with those used in the FSAR supporting previously approved licensing of Optimized Fuel Assembles.

While the DNBR margin will be reduced at the lower trip frequency, the results of the analysis indicate that the departure from nucleate boiling ratio (DNBR) remains well within the accepted limits. Therefore, the staff believes that the proposed amendments match the Commission's example of actions likely to involve no significant hazards considerations and proposes to determine that the amendments involve no significant hazards considerations.

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

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 1800 M Street NW., Washington, D.C. 20036.

NRC Branch Chief: James R. Miller.

Yankee Atomic Electric Company, Docket No. 50-29, Yankee Nuclear Power Station, Franklin County, Massachusetts

Date of amendment request: May 26, 1981 as revised January 23, 1984.

Description of amendment request: The proposed amendment would make a number of changes to the Technical Specifications (TSs). These changes are: (1) Correction of typographical errors and clarifications which do not change the intent of the TSs and involve no reduction in safety; (2) the addition of a further restriction in the TSs by removal of reference to 3 loop operation as 3 loop operations are not permitted at Yankee until further analysis is performed and approved by the NRC; (3) addition of limitations or restrictions in the TSs to be consistent with TMI Action Plan requirements of NUREG-0737 and NRC Generic Letter 82-16; (4) addition of limitations, restrictions or controls in the TSs to be consistent with NUREG-0825, "Integrated Plant Safety Assessment Systematic Evaluation Program for the Yankee Nuclear Power Station;" (5) revisions to radiological effluent TSs to correct typographical errors or achieve more consistency throughout the TSs; (6) revisions and additions to the TSs that involve additional restrictions or surveillance requirements in radiological effluent TSs to be more consistent with NUREG-0472, "Radiological Effluent Technical Specifications for PWRs;" (7) revision to the basis for TS 3/4.7.6, Sealed Source Contamination, to include a basis for exempting sealed sources contained within radiation monitoring or boron measuring devices from leak testing requirements. This will make the basis more consistent with other TSs and with Standard Technical Specifications. This change is administrative only and does not remove or relax any existing safety requirements as it affects the Basis of the TSs only; (8) revise TSs for Operational Quality Assurance Program which require conformance to ANSI N18.7-1972 and Regulatory Guide 1.33, of November 1972 to the more current NRC requirements stated in ANSI N18.7-1976 and in Regulatory Guide 1.33, Revision 2. In addition, this revision will be more consistent with the Yankee Operational Quality Assurance Program; and (9) revise TSs to reduce the time between Audits of the Facility Security Plan and Facility Emergency Plan from 24 months to 12 months. This reduction in time between audits is an additional restriction in the TSs. It is also more in conformance with Generic Letters 82-17 and 82-23 and the regulations of 10 CFR 50.54(t) and 10 CFR 73.40(d). These audits are conducted by the Nuclear Safety Audit and Review Committee.

The remaining issues addressed in the application dated May 26, 1981 as revised January 23, 1984 and February 26, 1985 will be addressed in future correspondence.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of standards for a no significant hazards consideraion by providing certain examples (48 FR 14870). The examples include: (i) A purely administrative change to the TSs to achieve consistency throughout the TSs, correct errors of clarify TSs; (ii) a change that constitutes an additional limitation, restriction or control not presently include in the TSs and (vii) a change to make the license conform to changes in the regulations.

Items (1) and (5) which correct typgraphical errors or achieve more consistency with other TSs are encompassed by the Commission's example (i) of actions not likely to involve a significant hazards consideration. Items (2), (3), (4), (6) and (9) which are additional limitations, restrictions or controls in the TSs are encompassed by the Commission's example (ii) of actions not likely to involve a significiant hazards consideration. Item (7) merely revises the basis for a Technical Specification. Item (8) is a change which would not: (1) Involve a significant increase in the probability or consequences of a 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 the margin of safety because the change does not significantly relax any existing requirements but updates the TS requirements to be consistent with the NRC approved Yankee Operational Quality Assurance Program. Therefore, Item (8) meets the standards provided in 10 CFR 50.92(c).

Therefore, Since Items (1) through (9) of the application for amendment involve proposed changes that are similar to examples or meet the standards for which no significant hazards consideration exists, the staff has made a proposed determination that the proposed amendment involves no significant hazards consideration.

Local Public Document Room location: Greenfield Community College, 1 College Drive, Greenfield, Massachusetts 01301.

Attorney for licensee: Thomas Dignan, Esquire, Ropes and Gray, 225 Franklin Street, Boston, Massachusetts 02107.

NRC Branch Chief: John A. Zwolinski.

PREVIOUSLY PUBLISHED NOTICES OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO OPERATING LICENSES AND PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION AND OPPORTUNITY FOR 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 because time did not allow the Commission to wait for this regular monthly notice. They are repeated here because the monthly notice lists all amendments 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.

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

Dates of application for amendment: February 1 and 15, 1985.

Brief description of amendment: The amendment revises the Technical Specifications to permit changes in the normal full power background trip level setting for the main steam line high radiation scram and isolation setpoints to accommodate a short-term test of operation with hydrogen injection into the reactor coolant.

Date of publication of individual notice in **Federal Register**: 50 FR 7860 February 26, 1985.

Expiration date of individual notice: March 27, 1985.

Local Public Document Room Location: Plymouth Public Library, North Street, Plymouth, Massachusetts 02360.

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

Date of amendment request: February 12, 1985.

Description of amendment request: The proposed amendment would allow for an extension of the time period for completion of the containment vessel tendon surveillances required by Technical Specification Surveillance 4.6.1.6.1.

Date of publication of individual notice in Federal Register: February 27, 1985 (50 FR 8024).

Expiration date of individual notice: March 28, 1985

Local Public Document Room locations: Fulton City Library, 709 Market Street, Fulton, Missouri 65251 and the Olin Library of Washington University, Skinker and Lindell Boulevards, St. Louis, Missouri 63130.

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

Date of application request: February 7, 1985.

Brief description of amendment: The amendment would provide relief from a restriction in the plant technical specifications on hydrotesting of the secondary side with the primary side above 350°F.

Date of publication of individual notice in **Federal Register**: March 4, 1985 (50 FR 8688).

Expiration date of individual notice: April 3, 1985.

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

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

During the 30-day period since publication of the last monthly 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 and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing in connection with these actions was published in the **Federal Register** as indicated. No request for a hearing or petition for leave to intervene was filed following this notice.

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 amendments, (2) the amendments, and (3) the Commission's related letters, Safety Evaluations and/or Environmental Assessments as indicated. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C., and at the local public document rooms for the particular facilities involved. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Alabama Power Company, Docket Nos. 50-348 and 50-364, Joseph M. Farley Nuclear Plant, Unit Nos. 1 and 2, Houston County, Alabama

Date of application for amendments: February 3, 1984 supplemented September 14, and November 26, 1984.

Brief description of amendments: Technical Specifications are modified to incorporate revisions in reporting requirements in response to Generic Letter 83-43 to comply with 10 CFR 50.72 and 50.73 which became effective January 1, 1984.

Date of issuance: February 19, 1985.

Effective date: February 19, 1985.

Amendment Nos.: 57 and 49.

Facilities Operating License Nos. NPF-2 and NPF-8. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 25, 1984 (49 FR 17851). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 19, 1985.

Significant hazards consideration comments received: No.

Local Public Document Room location: George S. Houston Memorial Library, 212 W. Burdeshaw Street, Dothan, Alabama 36303.

Arkansas Power and Light Company, Docket No. 50-313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of application for amendment: August 15, 1984.

Brief description of amendment: The amendment provides additional Technical Specifications for ANO-1 which require operating restrictions and testing of the Low Temperature Overpressure Protection System.

Date of issuance: March 4, 1985.

Effective date: March 4, 1985.

Amendment No.: 95.

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

Date of initial notice in **Federal Register**: September 28, 1984 (49 FR 38393). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 4, 1985.

No significant hazards consideration comments received: No.

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

Arkansas Power and Light Company, Docket Nos. 50-313 and 50-368, Arkansas Nuclear One, Units 1 and 2, Pope County, Arkansas

Date of application for amendments: October 16, 1984.

Brief description of amendment: The amendments revised the Technical Specifications to require that keys to key operated handswitches for the ANO 1 and 2 containment purge valves be removed when purge valves are required to be closed.

Date of issuance: March 18, 1985.

Effective date: March 18, 1985.

Amendment Nos.: 96 and 64.

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

Date of initial notice in **Federal Register**: November 21, 1984 (49 FR 45941 at 45943). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 18, 1985.

No significant hazards consideration comments received. No.

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

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

Date of application for amendments: April 9 and June 29, 1984.

Brief description of amendments: The amendments changed the Technical Specifications to provide Limiting Conditions for Operation and Surveillance Requirements for certain NUREG-0737 items.

Date of issuance: February 22, 1985.

Effective date: February 22, 1985.

Amendment Nos.: 99 and 81.

Facility Operating License Nos. DPR-53 and DPR-69. Amendments changed the Technical Specifications.

Date of initial notice in **Federal Register**: December 31, 1984 (49 FR 50794 at 50796). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 22, 1985.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: October 11, 1984.

Brief description of amendments: The amendments revised the Technical Specifications (TS) to: (1) Provide an environmental monitoring program which meets the requirements of Appendix I to 10 CFR Part 50, and (2) delete the existing environmental monitoring TS in the Appendix B TS which are no longer needed.

Date of issuance: February 22, 1985.

Effective date: February 22, 1985.

Amendment Nos.: 100 and 82.

Facility Operating License Nos. DPR-53 and DPR-69. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 31, 1984 (49 FR 50794 at 50799). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 22, 1985.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 29, 1984.

Brief description of amendments: The amendments changed Technical Specification 4.7.11.3c.2, "Halon Systems" to revise the Surveillance requirements for the Switchgear Room Halon and Cable Spreading Room total flood halon Fire suppression systems.

Date of issuance: March 7, 1985.

Effective date: March 7, 1985.

Amendment Nos.: 101 and 83.

Facility Operating License Nos. DPR-53 and DPR-69. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 31, 1984 (49 FR 50794 at 50797). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 7, 1985.

No significant hazards consideration comments received: No.

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

Carolina Power & Light Company, Docket No. 50-324, Brunswick Steam Electric Plant, Unit 2, Brunswick County, North Carolina

Date of application for amendment: September 26, 1984.

Brief description of amendment: The amendment changes the Technical Specifications by revising Tables 3.3.5.2-1 and 4.3.5.2-1 and TS 3/4.6.6.4 to reflect requirements for the drywell/suppression chamber hydrogen and oxygen analyzers. These requirements were identified in NUREG-0737 as TMI Action Plan Item II.F.1.6.

Date of issuance: February 20, 1985.

Effective date: February 20, 1985.

Amendment No.: 108.

Facility Operating License No. DPR-62. Amendment revised the Technical Specificatons.

Date of initial notice in **Federal Register**: November 2, 1984 (49 FR 45943). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 20, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Southport, Brunswick County Library, 109 W. Moore Street, Southport, North Carolina 28461.

Carolina Power and Light Company, Docket No. 50-261, H.B. Robinson Stem Electric Plant, Unit No. 2, Darlington, South Carolina

Date of application for amendment: March 21, 1984, as supplemented by November 8, 1984.

Brief description of amendment: The amendment revised the Technical Specification to provide conformance with 10 CFR 50.72 and 50.73 The licensee's second submittal dated November 8, 1984 was largely due to Amendments 83, 84, and 85 issued subsequent to their March 21, 1984 submittal. The subsequent amendments affected pages of the reporting requirements as described in the licensee's November 8, 1984 forwarding letter. Minor changes of a clarification nature were also made as a result of the NRC review process. Therefore, no substantive changes were made by the licensee's November 8, 1984 resubmittal.

Date of issuance: March 15, 1985.

Effective date: March 15, 1985.

Amendment No.: 89.

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

Date of initial notice in **Federal Register**: April 25, 1984 (49 FR 17857). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 15, 1985.

Significant hazards consideration comments received: No.

Local Public Document Room Location: Hartsville Memorial Library, Home and Fifth Avenues, Hartsville, South Carolina 29535.

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

Date of application for amendments: March 15, 1984 as supplemented by a letter dated September 21, 1984.

Brief description of amendments: The amendments delete the Technical Specifications (TS) snubber tables, 3.6.1.a and 3.6.1.b and all references to them to reflect the guidance in Generic Letter 84-13. Additionally, TS Sections 3.6.I.2 and 3.6.I.4 were revised to remove any reference to the Torus Ring Header Snubber work which has been completed at both units. Section 4.6.2 and the Bases for Section 3.6.I are revised to remove limits on the type of functional testing performed on the snubbers. Finally, Section 4.6.I.2 and 4.6.I.4 and the Bases for 3.6.I are being revised to allow for velocity range tests as required by certain types of snubbers which were not used at the site until recently.

Date of issuance: February 27, 1985.

Effective date: February 27, 1985.

Amendment Nos.: 85 and 78.

Provisional Operating License No. DPR-19 and Facility Operating License No. DPR-25. The amendments revise the Technical Specifications.

Date of initial notice in **Federal Register**: December 31, 1984 (50 FR 50800). The Commission's related evaluation of the amendments is contained in a letter dated February 27, 1985.

No significant hazards consideration comments received: No.

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

Commonwealth Edison Company, Docket No. 50-265, Quad Cities Nuclear Power Station, Unit 2, Rock Island County, Illinois

Date of application for amendment: December 4, 1984.

Brief description of amendment: The amendment revises the Technical Specifications to allow a temporary increase in the Linear Heat Generation Rate (LHGR) from 13.4 to 15.0 kw/ft for certain Barrier Fuel Test Assemblies present in the Unit 2 core. This new limit applies only during the remainder of the current Operating Cycle 7.

Date of issuance: February 25, 1985.

Effective date: February 25, 1985.

Amendment No.: 85.

Facility Operating License No. DPR-30. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 23, 1985, 50 FR 3049.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 25, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Moline Public Library, 504-17th Street, Moline, Illinois 61265.

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

Date of application for amendments: October 17, 1984 and supplemented January 3, and January 16, 1985.

Brief description of amendments: These amendments add a specification for hydrogen monitors to match the Standardized Technical Specifications and eliminate specifications for the hydrogen purge fan system which now serves only as a backup to the new operable hydrogen recombiner system.

The licensee's submittals of January 3 and 16, 1985 were made as a result of NRC staff request to clarify the language of the original submittal dated October 17, 1984, and do not contain substantive changes.

Date of issuance: March 14, 1985.

Effective date: March 14, 1985.

Amendment Nos.: 87 and 77.

Facility Operating License Nos. DRP-39 and DPR-48. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 31, 1984 (49 FR 50801).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 14, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Zion Benton Library District, 2600 Emmaus Avenue, Zion, Illinois 60099.

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

Date of application for amendment: February 28, 1984.

Brief description of amendment: The amendment revises the Technical Specifications to incorporate the requirements of NUREG-0737 Item II.B.1, "Reactor Coolant System Vents." The Technical Specifications have been revised to ensure that the Indian Point Unit No. 2 Reactor Coolant Vent System is available to effectively vent noncondensible gases from teh reactor coolant system without significantly increasing the probability of a Loss of Coolant Accident or challenge to containment integrity.

Date of issuance: February 28, 1985.

Effective date: Immediately with implementation within 30 days.

Amendment No.: 93.

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

Date of initial notice in **Federal Register**: April 25, 1984 (49 FR 17858) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 28, 1985.

Significant hazards consideration comments received: None.

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

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: July 31, 1984.

Brief description of amendments: The amendments change the Technical Specifications to expand Tables 3.3-10 and 4.3-7 concerning accident monitoring instrumentation and surveillance requirements to include the recently installed Reactor Vessel Level Instrumentation System and to include both channels of the Subcooling Margin Monitoring System.

Date of issuance: February 28, 1985.

Effective date: February 28, 1985.

Amendment Nos.: 40 and 21.

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

Date of initial notice in **Federal Register**: October 24, 1984 (49 FR 42817). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 28, 1985.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 10, 1984.

Brief description of amendment: The amendment changes the Technical Specifications for Beaver Valley Unit No. 1 to eliminate the Tables listing all mechanical and hydraulic snubbers, to add a new surveillance requirement on the recirculation spary subsystem, and to clarify a number of existing specifications.

Date of issuance: February 22, 1985.

Effective date: February 22, 1985.

Amendment No.: 91.

Facility Operating License No. DPR-66. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 31, 1984 (49 FR 50804). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 22, 1985.

No significant hazards consideration comments received. None.

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

Florida Power and Light Company, et al., Docket No. 50-389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of application of amendment: September 28 and October 19, 1984.

Brief description of amendment: The amendment changed the nomenclature of three valves in Tables 3.6-1 and 3.6-2.

Date of Issuance: March 15, 1985.

Effective Date: March 15, 1985.

Amendment No.: 10.

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

Date of initial notice in **Federal Register**: October 29, 1984 (49 FDR 43517). The Commission's related evaluation of the amendment is contained in a Safety Evaluation date March 15, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Ft. Pierce, Florida.

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50-321, Edwin I. Hatch Nuclear Plant, Unit No. 1, Appling County, Georgia

Date of application for amendment: February 6, 1984, as supplemented April 16, 1984.

Brief description of amendment: The amendment revises the TSs for Hatch Unit 1 to increase the reactor pressure operability requirement for the High Pressure Coolant Injection and Reactor Core Isolation Cooling systems from 113 psig to 150 psig.

Date of issuance: March 12, 1985.

Effective date: March 12, 1985.

Amendment No.: 107.

Facility Operating License No. DPR-57. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 20, 1984 (49 FR 25361). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 12, 1985.

No significant hazards consideration comments received: No.

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

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 13, 1984.

Brief description of amendment: This amendment revises the Technical pecifications to add operability and urveillance requirements for the ackup incore thermocouple display hannels in response to the equirements of NUREG-0737, Item F.2(8).

Date of issuance: March 5, 1985.

Effective date: 30 days after issuance.

Amendment No.: 105.

Facility Operating License No. DPR- Amendment revised the Technical ecifications.

Date of initial notice in **Federal** gister: April 25, 1984 (49 FR 17863).

Commission's related evaluation of amendment is contained in a Safety luation dated March 5, 1985.

o significant hazards consideration ments received: No.

cal Public Document Room tion: Government Publications on, State Library of Pennsylvania, ation Building, Commonwealth and ut Streets, Harrisburg, sylvania 17126.

Electric Light and Power Company, et No. 50-331, Duane Arnold y Center, Linn County, Iowa

e of application for amendment: nber 26, 1984, as supplemented er 26, 1984.

f description of amendment: The ment revises the license to the commitment of the use, by urity Force, of rifles from the Arnold Energy Center Security d the Guard Training and cation Plan. We are also updating nse to incorporate any § 50.54(p) that have occurred.

of issuance: February 26, 1985.

ive date: February 26, 1985.

dment No.: 112.

y Operating License No. DPR- ndment revised the license.

f initial notice in **Federal** November 21, 1984 (49 FR letter dated October 26, 1984, ee submitted a page correction tember 26, 1984 application. ittal falls within the scope of al notice.

nmission's related evaluation ndment is contained in a luation dated February 26,

ficant hazards consideration received: No.

blic Document Room dar Rapids Public Library, venue, SE., Cedar Rapids,

Iowa Electric Light and Power Company, Docket No. 50-331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: August 20, 1984, as revised September 14, 1984.

Brief description of amendment: This amendment revises the Technical Specification to: (1) Change the snubber testing following a failure from 10% to 5%, (2) delete the requirement to increase the drag force by 50% during snubber functional tests, (3) delete snubbers list from Technical Specifications, and (4) correct some typographical errors.

Date of issuance: March 12, 1985.

Effective date: March 12, 1985.

Amendment No.: 113.

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

Date of initial notice in **Federal Register:** November 21, 1984 (49 FR 45954). The September 14, 1984 submittal contained clarifying information and did not change the substance of the initial application, therefore, no additional notice was issued.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 12, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Cedar Rapids Public Library, 426 Third Avenue, SE., Cedar Rapids, Iowa 52401.

Iowa Electric Light and Power Company, Docket No. 50-331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: February 27, 1984, as revised August 16, 1984.

Brief description of amendment: This amendment revises the Technical Specifications to incorporate changes related to: (1) Clarification in 12 areas, (2) updating references in 12 areas, and (3) correction of typographic errors in five areas of the Technical Specifications.

Date of issuance: March 14, 1985.

Effective date: March 14, 1985.

Amendment No.: 114.

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

Date of initial notice in **Federal Register:** October 24, 1984 (49 FR 42824). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 14, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Cedar Rapids Public Library, 426 Third Avenue, SE., Cedar Rapids, Iowa 52401.

Niagara Mohawk Power Corporation, Docket No. 50-220, Nine Mile Point Nuclear Station, Unit No. 1, Oswego County, New York

Date of application for amendment: July 19, 1984.

Brief description of amendment: The amendment revises the Technical Specification to make changes to the Administrative Controls section to reflect a revised arrangement of certain upper management positions in the corporate organization.

Date of issuance: February 19, 1985.

Effective date: February 19, 1985.

Amendment No.: 68.

Facility Operating License No. DPR-63. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** September 28, 1984 (49 FR 38403). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 19, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: State University College at Oswego, Penfield Library—Documents, Oswego, New York 13126.

Niagara Mohawk Power Corporation, Docket No. 50-220, Nine Mile Point Nuclear Station, Unit No. 1, Oswego County, New York

Date of application for amendment: August 7, 1984.

Brief description of amendment: The amendment revised the Technical Specifications concerning changes to Section 6.5.2.8 of the Administrative Controls Section.

Date of issuance: February 25, 1985.

Effective date: February 25, 1985.

Amendment No.: 69.

Facility Operating License No. DPR-63. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** September 28, 1984 (49 FR 38403). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 25, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: State University College at Oswego, Penfield Library—Documents, Oswego, New York 13126.

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Public Service Electric and 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: December 27, 1983 and supplemented February 25, 1985.

Brief description of amendments: The amendments consists of three (3) independent parts. Part (1) modifies the Salem Unit 1 Technical Specifications, Table 3.3-1 (Action 1) and Table 3.3-3 (Action 13) to read the same as Salem Unit 2 Technical Specifications Tables 3.3-1 and 3.3-3. Part (2) corrects a typographical error in the Salem Unit 32 Technical Specifications. Part (3) revises the response time requirement for the overtemperature delta T reactor trip for both Units 1 and 2 and makes them identical.

The licensee's supplemental submittal dated February 25, 1985 provided an additional Westinghouse analysis which was done subsequent to the original license change request. This submittal will be the subject of a future action.

Date of issuance: March 8, 1985.

Effective date: March 8, 1985.

Amendment Nos.: 60 and 31.

Facility Operating Licenses Nos. DPR-70 and DPR-75: Amendments revised the Technical Specification.

Date of initial notice in **Federal Register**: December 31, 1984 (49 FR 50820). The Commission's related evaluation of amendments is contained in a Safety Evaluation dated March 8, 1985.

No significant hazards consideration comments have been received.

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

Rochester Gas and Electric Corporation, Docket No. 50-244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of application for amendment: January 25, 1985.

Brief description of amendment: The amendment modified the Technical Specifications authorizing the use of a temporary closure plate in place of the equipment hatch (door) during refueling operations.

Date of issuance: March 8, 1985.

Effective date: March 8, 1985.

Amendment No.: 2.

Facility Operating License No. DPR-18. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: February 5, 1985 (50 FR 5020).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 8, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610.

Sacramento Municipal Utility District, Docket No. 50-312, Rancho Seco Nuclear Generating Station, Sacramento County, California

Date of application for amendment: July 22, 1983, as supplemented June 26 and October 1, 1984.

Brief description of amendment: The amendment revises the Technical Specifications defining the requirements for surveillance of Auxiliary Feedwater System Auto-Start Instrumentation.

Date of issuance: February 21, 1985.

Effective date: February 21, 1985.

Amendment No.: 60.

Facility Operating License No. DPR-54. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: May 23, 1984 (49 FR 21837 and December 31, 1984, 49 FR 50824). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 21, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Sacramento City-County Library, 828 I Street, Sacramento, California.

Sacramento Municipal Utility District, Docket No. 50-312, Rancho Seco Nuclear Generating Station, Sacramento County, California

Date of application for amendment: January 26, 1984, as supplemented July 11, 1984, and revised October 30, 1984.

Brief description of amendment: The amendment revises the Technical Specifications to clarify the use of the term "Operable" as it applies to single-failure criterion for safety systems.

Date of issuance: March 4, 1985.

Effective date: March 4, 1985.

Amendment No.: 61.

Facility Operating License No. DPR-54. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 31, 1984 (49 FR 50822).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 4, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Sacremento City-County Library, 828 I Street, Sacramento, California.

Sacramento Municipal Utility District, Docket No. 50-312, Rancho Seco Nuclear Generating Station, Sacramento County, California

Date of application for amendment: June 25, 1984.

Brief description of amendment: The amendment revises the Technical Specifications to prescribe requirements for reporting operational conditions and events in accordance with 10 CFR 50.73.

Date of issuance: March 8, 1985.

Effective date: March 8, 1985.

Amendment No.: 63.

Facility Operating License No. DPR-54. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 31, 1984 (49 FR 50823).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 8, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Sacramento City-County Library, 828 I Street, Sacramento, California.

Sacramento Municipal Utility District, Docket No. 50-312, Rancho Seco Nuclear Generating Station, Sacramento County, California

Date of application for amendment: April 19, 1983, as supplemented November 14, 1983, and June 25, 1984.

Brief description of amendment: The amendment revises the Technical Specifications to describe the current off-site and on-site organizations and review committee membership and quorum requirements.

Date of issuance: March 7, 1985.

Effective date: March 7, 1985.

Amendment No.: 62.

Facility Operating License No. DPR-54. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 25, 1984 (49 FR 17872 and December 31, 1984, 49 FR 50823). The Commission's related evaluation of the amendment is contained in a Safety evaluation dated March 7, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Sacramento City-County Library, 828 I Street, Sacramento, California.

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

Date of application for amendment: July 17, 1984, as revised on November 30, 1984.

Brief description of amendment: The amendment modifies the Technical Specifications by adding administrative guidance and requirements relating to the assignment of overtime to personnel performing safety-related activities.

Date of issuance: March 6, 1985.

Effective date: March 6, 1985.

Amendment No.: 88.

Provisional Operating License No. DPR-13. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 23, 1985 (50 FR 3055). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 6, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: San Clemente Public Library, 242 Avendia Del Mar, San Clemente, California 92672.

Southern California Edison Company, et al. Docket Nos. 50-361 and 50-362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California

Dates of application for amendments: February 29, April 2, July 2, August 7, October 1 and 3, 1984.

Brief description of amendments: The amendments changes Technical Specifications to: (1) Accommodate Core Protection Calculator software changes being implemented for Cycle 2 operation, (2) allow Control Element Assembly misalignment during requried physics testing, (3) be consistent with the assumptions used for Cycle 2 safety analysis, (4) incorporate the results of the revised departure from nucleate boiling ration (DNBR) analysis and explicitly define the actions required if the core operating limit supervisory system is out-of-service and one or both control element assembly calculators are inoperable, and (5) change certain specifications relating to reactor protection instrumentation and electrical power sources.

Date of issuance: March 1, 1985.

Effective date: Portions of the amendments are effective as of the date of issuance and shall be fully implemented within 30 days; the remainder of the amendments is effective on initial entry into the first applicable operational mode following first refueling.

Amendment Nos.: 32 and 21.

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

Dates of initial notices in **Federal Register:** November 21, 1984 (49 FR 45964 and 45965 and 45966) and December 31, 1984 (49 FR 50843 and 50845). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 1, 1985.

No significant hazards consideration comments were received.

Local Public Document Room location: San Clemente Library, 242 Avenida Del Mar, San Clemente, California.

The Toledo Edison 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: May 5, 1982 (Part of Item 1), and March 22, 1984.

Brief description of amendment: The amendment modifies Tables 3.3-10 and 4.3-10 relating to post-accident monitoring instrumentation by adding incore thermocouples, reactor coolant hot leg level, containment water level, and containment pressure to the list of post-accident instrumentation that must be operable and are subject to surveillance requirements. The amendment also adds TS Section 6.8.4.c which requires the establishment of a post-accident sampling program.

Date of issuance: March 13, 1985.

Effective date: March 13, 1985.

Amendment No.: 84.

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

Date of initial notice in **Federal Register:** October 24, 1984 (49 FR 42835). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 13, 1985.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 1, 1984.

Brief description of amendment: The amendment consists of a change to the Technical Specifications to add an additional provision to allow for appropriate compensatory actions when two range monitor channels are out of sevice in order to maintain the plant in a safe condition.

Date of issuance: March 6, 1985.

Effective date: March 6, 1985.

Amendment No.: 4.

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

Date of initial notice in **Federal Register:** September 28, 1984 (50 FR 38413). The Commission's related evaluation of the amendment is contained in a Safety Evaluation, dated March 6, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Fulton City Library, 709 Market Street, Fulton, Missouri 65251 and Olin Library of Washington University, Skinker and Lindell Boulevard, St. Louis, Missouri 63130.

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

Date of application for amendment: March 26, 1984, as supplemented September 9, 1984.

Brief description of amendment: The amendment revises the Technical Specifications to reflect a change from 120% to 140% in the main steam line high flow setpoint. In addition, the reactor power limit for quarterly MSIV full closure testing in increased from 50% to 75% of rated power.

Date of issuance: February 21, 1985.

Effective date: February 21, 1985.

Amendment No.: 86.

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

Date of initial notice in **Federal Register:** May 23, 1984 (49 FR 21848). By letter dated September 7, 1984, the licensee submitted clarifying information which falls within the scope of the initial notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation date February 21, 1985.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 8, 1984.

Brief description of amendments: The amendments revised Technical

Specification 15.3-10 to define the "fully withdrawn" condition of a control rod as equal to or greater than 225 steps withdrawn. Figure 15.3.10-1 "Control Rod Insertion Limits" has been revised to change "steps withdrawn" to "percentage fo control bank withdrawn".

Date of issuance: March 7, 1985.

Effective date: 20 days after issuance.

Amendment Nos.: 88 and 93.

Facility Operating License Nos. DPR-24 and DPR-27:

Date of initial notice in Federal Register: September 28, 1984 (49 FR 38390 at 38414). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 7, 1985.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: May 2, 1984 as revised September 5, 1984.

Brief description of amendments: The amendments revised the surveillance requirements for containment prestressed tendons and added a limiting condition for operation.

Date of issuance: March 7, 1985.

Effective date: 20 days from date of issuance.

Amendment Nos.: 89 and 94.

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

Date of initial notice in Federal Register: June 20, 1984 (49 FR 25350 at 25382). Renoticed November 21, 1984 (49 FR 45941 at 45980). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 7, 1985.

No significant hazards consideration comments received: No.

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

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND FINAL DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION AND OPPORTUNITY FOR HEARING (EXIGENT OR EMERGENCY CIRCUMSTANCES)

During the 30-day period since publication of the last monthly 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. 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 and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing. For exigent circumstances, a press release seeking public comment as to the proposed no significant hazards consideration determination was used, and the State was consulted by telephone. In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant, a short public comment period (less than 30 days) has been offered and the State 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, 1717 H Street, NW., Washington, D.C., and at the local public document room for the particular facility involved.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendments. By April 26, 1985, 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 petition for leave to intervene. Requests for a hearing and petitions 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. 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 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 fifteen (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 fifteen (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, and the bases for each contention set forth with reasonable specificity. Contentions shall be limited to matters within the scope of the amendment under consideration. 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, D.C. 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C., by the above date. Where petitions are filed during the last ten (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 (800) 325-6000 (in Missouri (800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to *(Branch Chief)*: 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 Executive Legal Director, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board designated to rule on the petition and/or request, that the petitioner has made a substantial showing of good cause for the granting of a late petition and/or request. That determination will be based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

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

Date of application for amendment: December 6, 1984, as supplemented January 10, 1985, February 8, 14, and 21, 1985.

Brief description of amendment: The amendment revises the Technical Specifications to permit refueling with the Reactor Protection System and certain specified refueling interlocks and control rod blocks inoperable. These revisions will facilitate installation of Analog Trip Transmitter components during the Reload 6/Cycle 7 refueling outage.

Date of issuance: February 22, 1985.

Effective date: February 22, 1985.

Amendment No.: 87.

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

Public comments requested as to proposed no significant hazards consideration: Yes. February 4, 1985 (50 FR 4929).

No comments received.

The Commission's related evaluation of the amendment and final determination of no significant hazards consideration are contained in a Safety Evaluation dated February 22, 1985.

Local Public Document Room location: Penfield Library, State University College of Oswego, Oswego, New York.

Dated at Bethesda, Maryland this 20th day of March 1985.

For the Nuclear Regulatory Commission.

James R. Miller,

Chief, Operating Reactors Branch #3, Division of Licensing.

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