123 Main Street
White Plains, New York 1[illegible]
914 681.6240

New York Power Authority

June 23, 1989
IPN-89-035

John C. Brons
Executive Vice President
Nuclear Generation

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Detailed Control Room Design Review

References:

1. Letter from Mr. John C. Brons to Mr. Steven A. Varga, dated October 31, 1985, entitled: "Detailed Control Room Design Review Summary Report."

2. Letter from Mr. John C. Brons to Mr. Steven A. Varga, dated January 7, 1986, entitled: "Regulatory Guide 1.97 Implementation Program."

3. Letter from Mr. John C. Brons to Mr. Steven A. Varga, dated September 30, 1986, entitled: "Detailed Control Room Design Review."

4. Letter from Mr. John C. Brons to Mr. Steven A. Varga, dated December 18, 1986, entitled: "Detailed Control Room Design Review."

5. Letter from Mr. John C. Brons to the NRC, dated November 2, 1987, entitled: "Detailed Control Room Design Review."

Dear Sir:

Refererence (1) submitted the two volume Indian Point 3 (IP-3) Detailed Control Room Design Review (DCRDR) Summary Report. Included in this report was a summary of the DCRDR methodology, a description of the human factors maintenance program to be implemented at IP-3 and a summary and resolution for each identified human engineering deficiency (HED). References (3), (4) and (5) provided supplemental information to the NRC concerning DCRDR for IP-3. Reference (2) submitted the Authority's evaluation of Regulatory Guide 1.97 with regard to IP-3. Enclosures B and C of Reference (2) contained the Authority's compliance survey and notes for Regulatory Guide 1.97, Revision 3. As part of these enclosures the Authority committed to address instrumentation identification in conjunction with the DCRDR program in order to provide a coordinated human factors approach. The purpose of this letter is to provide the Authority's position, from a human factors approach, on

A003
1/1

instrumentation identification (Regulatory Guide 1.97 Revision 3, Table 1, Item 8 - Equipment Identification).

The Authority, in accordance with Regulatory Guide 1.97, Revision 3, Table 1, Item 8 - Equipment Identification, identified all the instruments that might require some type of markings on the control panels so the operator could easily discern that they are intended for use under accident conditons. The Authority compared the list of instruments with the Emergency Operating Procedures (EOPs) and it was noted that the EOPs utilize instruments that are on the list of Regulatory Guide 1.97 required instruments as well as instruments that are not. The EOPs contain all of the instruments necessary for the safe operation of IP-3 during and after an emergency as well as instruments necessary to protect equipment which may or may not be safety related. Marking the Regulatory Guide 1.97 instrumentation would not be consistent with the Authority's objectives. The operators are well trained in the existing EOPs and marking various instruments will only cause confusion for the operators and would create a human factor concern.

It is the Authority's position that identifying instruments on the control panels with some sort of markings is unnecessary and will only cause confusion for the operators. It would not enhance the safe operation of IP-3 and in fact may reduce safety. Attachment I to this letter contains this item as an HED with its resolution for your review.

Should you or your staff have any questions concerning this matter, please call Mr. P. Kokolakis of my staff.

Very truly yours,

John C. Brons
Executive Vice President
Nuclear Generation

cc: Mr. Joseph D. Neighbors, Sr. Proj. Mgr.
U.S. Nuclear Regulatory Commission
Mail Stop 14B2
Washington, D.C. 20555

Resident Inspector's Office
U.S. Nuclear Regulatory Commission
P.O. Box 337
Buchanan, NY 10511

U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

ATTACHMENT I TO IPN-89-035
EQUIPMENT IDENTIFICATION HED

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DPR-64

HED - Types A, B and C instruments designated as Categories 1 and 2 in accordance with Regulatory Guide 1.97 Revision 3, are not identified with a common designation on the control panels so that the operator can easily discern that they are intended for use under accident conditions.

Resolution - Investigation by the Authority has determined that marking all type A, B and C instruments designated as Categories 1 and 2 would actually cause confusion in the control room. Existing Emergency Operating Procedures (EOPs) identify some instruments to be relied on that would not be marked. The EOPs identify all of the instruments necessary for the safe operation of IP-3 during and after an emergency as well as instruments necessary to protect equipment which may or may not be safety related. Marking the Regulatory Guide 1.97 instrumentation would not be consistent with the Authority's objectives. It would also be impractical and unnecessary to mark all the instruments identified in the EOPs. This would cause confusion for the operator and may reduce the effectiveness of operator action. Therefore, no action is necessary.

DISTRIBUTION
Docket file w/encl.
PDI-1 Rdg
CVogan
DNeighbors
RCapra
JScinto

June 14, 1989

DOCKET NO(S). 50-286

Mr. John C. Brons
Executive Vice President, Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

SUBJECT: **POWER AUTHORITY OF THE STATE OF NEW YORK**
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

The following documents concerning our review of the subject facility are transmitted for your information.

- [ ] Notice of Receipt of Application, dated __________.
- [ ] Draft/Final Environmental Statement, dated __________.
- [ ] Notice of Availability of Draft/Final Environmental Statement, dated __________.
- [ ] Safety Evaluation Report, or Supplement No. __________ dated __________.
- [ ] Environmental Assessment and Finding of No Significant Impact, dated __________.
- [ ] Notice of Consideration of Issuance of Facility Operating License or Amendment to Facility Operating License, dated __________.
- [x] Bi-Weekly Notice; Applications and Amendments to Operating Licenses Involving No Significant Hazards Considerations, dated May 31, 1989 [see page(s)] June 30, 1989.
- [ ] Exemption, dated __________.
- [ ] Construction Permit No. CPPR-__________, Amendment No. __________ dated __________.
- [ ] Facility Operating License No. __________, Amendment No. __________ dated __________.
- [ ] Order Extending Construction Completion Date, dated __________.
- [ ] Monthly Operating Report for __________ transmitted by letter dated __________.
- [ ] Annual/Semi-Annual Report- __________ transmitted by letter dated __________.

Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: **See next page**

MA 4

OFFICE	PDI-1						
SURNAME	CVogan cw						Cc
DATE	6/14/89						

NRC FORM 318 (10/80) NRCM 0240

OFFICIAL RECORD COPY

cc:

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Mr. Gerald C. Goldstein
Assistant General Counsel
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

Mr. Phillip Bayne, President
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Mr. William Josiger
Resident Manager
Indian Point 3 Nuclear Power Plant
Post Office Box 215
Buchanan, New York 10511

Mr. George M. Wilverding, Manager
Nuclear Safety Evaluation
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Director, Technical Development
Programs
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Resident Inspector
Indian Point Nuclear Generating
U.S. Nuclear Regulatory Commission
Post Office Box 337
Buchanan, New York 10511

Mr. Robert L. Spring
Nuclear Licensing Engineer
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Mr. A. Klausmann, Vice President
Quality Assurance
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, New York 10511

Mr. F. X. Pindar
Quality Assurance Superintendent
Indian Point 3 Nuclear Power Plant
Post Office Box 215
Buchanan, New York 10511

Mr. R. Beedle, Vice President
Nuclear Support
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

cc

Mr. Peter Kokolakis, Director
Nuclear Licensing
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Ms. Donna Ross
New York State Energy Office
2 Empire State Plaza
16th Floor
Albany, New York 12223

Mr. S. S. Zulla, Vice President
Nuclear Engineering
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Vice President
Nuclear Operations
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Charlie Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, New York 10271

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from May 6, 1989 through May 19, 1989. The last biweekly notice was published on May 17, 1989 (54 FR 21297).

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.

Written comments may be submitted by mail to the Regulatory Publications Branch, Division of Freedom of Information and Publications Services, Office of Administration and Resources Management, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room P-216, Phillips Building, 7920 Norfolk Avenue, Bethesda, Maryland from 7:30 a.m. to 4:15 p.m. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for hearing and petitions for leave to intervene is discussed below.

By June 30, 1989 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 amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment 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 Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. Where petitions are filed during the last 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 1-(800) 325-6000 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to (*Project Director*): petitioner's name and telephone number; date petition was mailed; plant name; and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

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

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

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

Date of application for amendments: June 22, 1983

Description of amendment request: The amendments would delete the residual heat removal (RHR)/service water discharge differential pressure instrument (transmitter and indicator) from the Technical Specifications (TS) for each unit. The licensee states that operability is not required to ensure the RHR and service water systems function as designed. The subject indication is located on the remote shutdown panel for each unit. This panel is utilized to shut down the unit and maintain shutdown conditions in the event control room habitability is lost.

Basis for proposed no significant hazard consideration determination: The Commission has provided standards for determining whether a no significant hazard consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license 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 Carolina Power & Light Company (CP&L) has reviewed the proposed changes to the TS and has determined that the requested amendment does not involve a significant hazards consideration for the following reasons:

1. The design, function, and operation of the plant systems will remain unchanged. Item 9, "Residual Heat Removal Service Water Discharge Differential Pressure," and instruments E11-PDT-N002BX and E11-PDI-3344 are being deleted from Tables 3.3.5.2-1 and 4.3.5.2-1 because they are not required to ensure that the RHR and service water systems function as designed. The RHR heat exchanger differential pressure is not an indicator of heat exchanger performance; therefore, it does nothing to ensure that sufficient capability is available to permit shutdown and maintenance of hot shutdown from locations outside the control room. Thus, these instruments should not be considered remote shutdown monitoring instruments and should be deleted from Tables 3.3.5.2-1 and 4.3.5.2-1. Since they are not remote shutdown monitoring instruments, their deletion from Table 3.3.5.2-1 and 4.3.5.2-1 will not increase the probability of an accident, nor will it change the consequences of an accident previously evaluated.

2. Instruments E11-PDT-N002BX and E11-PDI-3344 provide relative system pressure only and are not relied upon for remote shutdown. Thus, these instruments need not be considered remote shutdown monitoring instruments and should be removed from Tables 3.3.5.2-1 and 4.3.5.2-1. Their deletion will not create the possibility of a new or different kind of accident because they were not relied upon under accident conditions for remote shutdown purposes and they will continue to perform their design function in the same manner as before.

3. The capability to permit shutdown and maintain hot shutdown of the facility from outside the control room is not compromised by deleting Instruments E11-PDT-N002BX and E11-PDI-3344 from Tables 3.3.5.2-1 and 4.3.4.2-1. These instruments currently do not perform a remote shutdown monitoring function; they only provide a relative system differential pressure which is not relied upon in the BSEP remote shutdown procedures to achieve remote shutdown. Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

The staff has reviewed the CP&L determinations and is in basic agreement with them. RHR/Service Water differential pressure indication on the remote shutdown panel is not required to shut the plant down or keep it shut down; and this indication is not utilized in the procedure entitled "Plant Shutdown from Outside Control Room." Accordingly, the Commission proposes to determine that these changes do not involve a significant hazards consideration.

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

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

NRC Project Director: Elinor G. Adensam

Connecticut Yankee Atomic Power Company, Docket No. 50-213, Haddam Neck Plant, Middlesex County, Connecticut

Date of amendment request: April 14, 1989

Description of amendment request: The proposed amendment provides changes to Technical Specification Sections 3.17.1, "Axial Offset" and 3.17.2, "Linear Heat Generation Rate" to support coastdown operation of the Haddam Neck Plant at the end of Cycle 15.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exist as stated in 10 CFR 50.92. The licensee has evaluated the proposed amendment against the standards provided in 50.92 and determined that the proposed amendment will not:

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

Operation during coastdown is beyond the scope of the current design basis LOCA analysis for the Haddam Neck Plant. The current design basis assumes operation consistent with the Tave program. Coastdown operation immediately following end of core life maintains 100% thermal power by reducing Tave, while fully opening the turbine control valves. Post LOCA analysis sensitivities have shown that a reduction in the core inlet temperature, while maintaining full power, yields an increase in the projected peak cladding temperature (PCT) for the large break LOCA. This increase in PCT is due to the reduction in reverse core flow during blowdown after the coastdown of the reactor coolant pumps. The reduced flow yields higher temperatures at the end of blowdown, which yields a higher PCT after the adiabatic heatup and beginning of core recovery.

These sensitivities have also shown that between 100 and 90% power, the sensitivity to the core inlet temperature becomes insignificant relative to the drop in power.

In order to bound operation during coastdown, the current limiting case (double ended, cold leg guillotine, $C_D = 1.0$) was re-analyzed at full power, but with a bounding coastdown core inlet temperature at 90% power (Tinlet = 510° F). This re-analysis shows that the limiting LHGR must be reduced from the normal end of cycle value of 14.6 kw/ft to 13.5 kw/ft to maintain the PCT less than the Interim Acceptance Criteria limit of 2300° F.

The axial offset (AO) limits were developed for the new LHGR limit during coastdown. The new AO limits are slightly more restrictive on the negative side.

The proposed changes ensure that there is no increase or change in the probability of occurrence of any design basis accidents.

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

The proposed changes to the technical specifications ensure that the plant response to an accident during coastdown operation is essentially within the design basis of the plant.

The large break LOCA response during coastdown has been changed, but the PCT as a function of time after break retains the key characteristics associated with blowdown, refill and reflood. The reduction in the LHGR ensures that the PCT remains less than 2300° F for a postulated design basis event during coastdown conditions.

There are no new failure modes associated with the proposed technical specification changes. Therefore, the changes do not present the possibility for a new, unanalyzed accident.

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

The proposed changes to the technical specifications ensure that the margins of safety have not been reduced significantly. The reduction in the allowable LHGR from 14.6 to 13.5 kw/ft during coastdown operation after the end of core life restores the PCT to less than 2300° F. The new AO limits provide alarm points to assure that operation above an LHGR of 13.5 kw/ft is prohibited. Since the PCT remains less than 2300° F, there is no impact on the protective boundaries.

The NRC staff has reviewed this analysis and based on this review, it appears that the three criteria are satisfied. Therefore, the NRC staff proposes to determine that the application for amendment involves no significant hazards considerations.

Local Public Document Room location: Russel Library, 123 Broad Street, Middletown, Connecticut 06457.

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

NRC Project Director: John F. Stolz

Connecticut Yankee Atomic Power Company, Docket No. 50-213, Haddam Neck Plant Middlesex County, Connecticut

Date of amendment request: April 21, 1989

Description of amendment request: The proposed amendment will revise and combine Technical Specification Section 3.6, "Core Cooling Systems," Section 3.7, "Minimum Water Volume and Boron Concentration in the Refueling Water Storage Tank," and Section 4.3, "Core Cooling Systems - Periodic Testing," into a new Technical Specification Section 3.6 titled "Emergency Core Cooling Systems." This new section will follow the Westinghouse Standard format Technical Specifications.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. In accordance with 10 CFR 50.92 Connecticut Yankee Atomic Power Company has reviewed the proposed Technical Specification and concluded that they do not involve a significant hazards consideration because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated. The determination of whether or not a proposed change is equivalent, more restrictive (or a new requirement) or less restrictive is based on the Limiting Condition for Operation (LCO) and Applicability Requirements since it is these requirements which will impact the design basis accidents. In general, the conversion to the W STS yields more extensive and/or restrictive Action or Surveillance Requirements. As described above, most of the changes are more restrictive in that there are no comparable requirements in the existing Technical Specifications, and the proposed changes are equivalent to the W STS. For the few changes that are less restrictive, justification is provided for the changes. Since the proposed Sections 3.6.1 through 3.6.4 do not reduce the availability or the reliability of the ECCS, the consequences of the design basis accidents remain unchanged.

2. Create the possibility of a new or different kind of accident from any previously evaluated. Because there are no hardware modifications associated with the proposed changes, the performance of safety related systems remains unaffected during operations. The operability requirements are increased over the current requirements thus enhancing the performance of safety systems. Therefore, the proposed Technical Specifications will not modify the plant response to the point where it can be considered a new accident nor are any credible failure modes created.

3. Involve a significant reduction in a margin of safety. Because the changes proposed herein provide acceptable results for the design basis accident, no additional burden will be placed on the protective boundaries for postulated accidents. In addition, there are no plant hardware modifications associated with this change and hence, there is no direct impact on the protective boundaries. The proposed Technical Specifications do not affect the safety limits of the protective boundaries and the bases of the proposed Technical Specifications have been modified to reflect the proposed changes.

The NRC staff has reviewed this analysis and based on this review, it appears that the three criteria are satisfied. Therefore, the NRC staff proposes to determine that the application for amendment involves no significant hazards considerations.

Local Public Document Room location: Russel Library, 123 Broad Street, Middletown, Connecticut 06457.

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

NRC Project Director: John F. Stolz

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

Date of amendment request: September 30, 1988, as supplemented January 10, 1989, March 30, 1989 and April 14, 1989.

Description of amendment request: This amendment would amend the Indian Point Unit No. 2 Operating License and Technical Specifications to authorize operation of the plant at core power levels not in excess of 3071.4 MWt. The following changes to the Operating License and Technical Specifications would be included in the proposed amendment:

1. Rated Power - The proposed amendment would increase the maximum allowable power level in license condition 2.C.1 of the Operating License and the definition of Rated Power in Technical Specification 1.1.a. from 2758 MWt to 3071.4 MWt.

2. Overtemperature Delta-T and Overpower Delta-T Setpoints - The proposed amendment would revise the nominal average temperature value at rated power for the Overtemperature Delta-T and Overpower Delta-T protection logic functions in Technical Specifications 2.3.1.B(4) and 2.3.1.B(5) from 570° F to less than or equal to 579.7° F. These changes would reflect the increased temperature allowed at the increased power level.

3. DNB Parameters - The proposed amendment would increase the allowable average reactor coolant system temperature in Technical Specification 3.1.G.a from less than or equal to 573.5° F to less than or equal to 587.2° F. This change would reflect the increased average temperature allowed at the increased power level.

4. Auxiliary Feedwater Flow - The proposed amendment would increase the minimum required flow capability of each of the auxiliary feedwater pumps as specified in Technical Specification 3.4.A.(2) from 300 gpm to 380 gpm.

5. Secondary Steam Flow - The proposed amendment would revise the Basis for Technical Specification 3.4 to reflect the increased steam flow (increased from 11,669,736 to 13,310,000 lbs/hr) that would be associated with operation at the increase power level. The percentage of total main steam safety valve relieving capacity that this increased steam flow represents would also be changed from 129 percent of total secondary steam flow to 114 percent of total secondary steam flow.

6. Decay Time Prior to Fuel Movement - The proposed amendment would increase the minimum time specified in Technical Specification 3.8.B.4 and its associated Basis required following plant shutdown before fuel may be handled in the reactor from 131 hours to 174 hours.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 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.

In accordance with the above criteria, the licensee provided the following no significant hazards analysis for the six categories of changes discussed above.

1. Rated Power

The analysis results show that the proposed changes do not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with these changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. This is based on the original design basis of the plant, as confirmed by the analyses supporting original plant licensure. These include an environmental evaluation which assumed stretch rated conditions and a radiological evaluation conducted at 3216 MWt. These analyses have been further confirmed by analyses performed pursuant to the methodology of WCAP-10263.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. This is based on the system and components reviews accompanying original plant licensure, as confirmed by analyses recently conducted, all of which verify the capability of systems and components to operate at the stretch rated conditions.

(3) Involve a significant reduction in a margin of safety. Accident analyses, both past and present, performed at the stretch rated conditions demonstrate that DNB design basis remains unchanged, that the RCS pressure limit of 2735 psig will not be exceeded, and that LOCA results remain well below the regulatory limits given in 10 CFR 50.46.

Based on the above, the licensee concludes that the amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated; does not create the possibility of a new or different kind of accident from any accident previously evaluated; and does not involve a reduction in a required margin of safety.

2. Overtemperature Delta-T and Overpower Delta-T Setpoints

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (51 FR 7751). Example (vi) of those involving no significant hazards consideration discusses a change which may reduce a safety margin but where the results are clearly within all acceptance criteria with respect to the system or component. The proposed change reflects the increased average temperature allowed at the increased power level.

The results of all analyses show that the proposed changes do not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with these changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revision is being supported by conservative evaluation and analyses utilizing the latest approved computer codes and methodology. These analyses demonstrate conformance to the applicable design and regulatory criteria.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes to the Overtemperature Delta-T and Overpower Delta-T setpoint functions for reactor trip do not modify the plant's configuration or operation, and therefore the identical postulated accidents are the only ones that require evaluation and resolution. Nothing would be added or removed that would conceivably introduce a new or different kind of accident mechanism or initiating circumstances than that previously evaluated.

In general, the proposed changes do not adversely affect the ability of Overtemperature Delta-T and Overpower Delta-T reactor trip signals to perform their safety function to initiate reactor core shutdown during an Overtemperature Delta-T or Overpower Delta-T transient condition, respectively.

(3) Involve a significant reduction in a margin of safety. With the proposed change, all safety criteria previously evaluated are met, remain conservative, and continue to maintain the previous margins of safety.

The safety function of reactor trip on Overtemperature Delta-T and Overpower Delta-T is to initiate reactor core shutdown during Delta-T transient events to ensure that the reactor core safety limits as defined in Technical Specification Figure 2.1-1 are not exceeded. Safety evaluations and analyses for all of the licensing basis accidents described in FSAR Chapter 14 which take credit for an Overtemperature Delta-T or Overpower Delta-T reactor trip have been performed and the results of these analyses demonstrate conformance with the applicable design and regulatory requirements.

3. DNB Parameters

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (51 FR 7751). Example (vi) of those involving no significant hazards consideration discusses a change which may reduce a safety margin but where the results are clearly within all acceptance criteria with respect to the system or component. The proposed change is to increase the allowable Reactor Coolant System average temperature at 100% power.

All analyses performed show that the proposed changes do not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with these changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The Tavg value represents a design limit for average Reactor Coolant System temperature. This proposed change is supported by conservative analyses and evaluations based on approved codes and methodologies. All applicable design and safety criteria continue to be satisfied.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change in the design value of Tavg does not modify the plant's configuration or operation, and therefore the identical postulated accidents are the only ones that require evaluation or resolution. Nothing would be added or removed that would conceivably introduce a new or different kind of accident mechanism or initiating circumstances than that previously evaluated.

(3) Involve a significant reduction in a margin of safety. With the proposed change, all safety criteria used in previous analyses are met, remain conservative, and continue to maintain the previous margins of safety. Approved analysis codes and methodologies were employed as the basis for evaluating this proposed change.

All applicable design and safety criteria are expected to be satisfied including the impact of an increased Tavg.

4. Auxiliary Feedwater Flow

Consistent with the Commission's criteria in 10 CFR 50.92, the licensee has determined that the proposed change does not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with this change would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revision is supported by conservative evaluation and analyses utilizing the latest approved computer codes and methodology. These analyses have demonstrated conformance to the applicable design and regulatory criteria.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change to the minimum auxiliary feedwater pump flowrate does not modify the plant's configuration or operation, and therefore the identical postulated accidents are the only ones that require evaluation and resolution. Nothing would be added or removed that would conceivably introduce a new or different kind of accident mechanism or initiating circumstances than that previously evaluated.

In general, the proposed change does not adversely affect the ability of the auxiliary feedwater system to perform its safety function to supply high pressure feedwater to the steam generators to maintain a water inventory.

(3) Involve a significant reduction in a margin of safety. With the proposed change, all safety criteria previously evaluated are still met, remain conservative, and continue to maintain the previous margins of safety.

The safety function of the auxiliary feedwater system is to supply high pressure feedwater to the steam generators to maintain a water inventory. Safety evaluation and analyses for all of the licensing basis accidents described in FSAR Chapter 14 which take credit for the auxiliary feedwater system have been performed and the results of these analyses and evaluation have demonstrated conformance with the applicable design and regulatory requirements.

5. Secondary Steam Flow

The Commission has provided guidance concerning the application to the standards for determining whether a significant hazards consideration exists by providing certain examples (51 FR 7751) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to 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.

The proposed changes are purely administrative changes to achieve consistency with the Technical Specifications, and consistency with the proposed increase in licensed NSSS power.

Consistent with the Commission's criteria in 10 CFR 50.92, the licensee has determined that the proposed changes described above do not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with these changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluate. The proposed revisions do not affect plant operations. The proposed revisions provide consistency with Technical Specifications associated with the proposed increase in licensed NSSS power.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not modify the plant's configuration or operation. Nothing would be added or removed that would conceivable introduce a new or different kind of accident mechanism or initiating circumstance than those previously evaluated.

(3) Involve a significant reduction in a margin of safety. With the proposed changes, all safety criteria previously evaluated are still met, remain conservative, and continue to maintain the previous margins of safety. Because these changes are administrative in nature their implementation does not affect any margin of safety.

6. Decay Time Prior to Fuel Movement

Consistent with the Commission's criteria in 10 CFR 50.92, the licensee has determined that the proposed change does not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with this change would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluate. The radiological consequences are unchanged from those previously evaluated. Only the time after shutdown before fuel can be handled has been increased. Hence, neither the probability nor the consequences of the accident have increased.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The postulated fuel handling accident is the same as that previously evaluated.

(3) Involve a significant reduction in a margin of safety. The decay time before fuel can be handled has been increased to ensure that the radiological consequences will be appropriately within the guidelines of 10 CFR Part 100. Hence, the margin of safety is unchanged.

The staff agrees with the licensee's analysis. Therefore, based on the above, the staff proposes that the proposed amendment will not involve a significant hazards consideration.

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

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

NRC Project Director: Robert A. Capra

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

Date of amendment request: September 23, 1988 supplemented by letter dated January 16, 1989.

Description of amendment request: The proposed amendment would revise the requirements of the Technical Specifications relating to administrative controls. The modifications are intended to strengthen both the offsite and onsite safety review functions. The proposed changes establish the Plant General Manager as Chairman of the Plant Review Committee (PRC), eliminate the Plant Safety Engineering function, and establish the Plant Safety and Licensing Department, and the Nuclear Safety Services Department.

Basis for proposed no significant hazards consideration determination: The Commission has made a proposed determination that the amendment involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.59, this means that the 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 licensee has evaluated the proposed changes against the above standards as required by 10 CFR 50.91(a). The Commission has reviewed the licensee's evaluation and agrees with it. The Commission concludes that:

A. The changes do not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)) because the changes are organizational and administrative in nature. The proposed changes merely affect the manner by which the safety review function is conducted. The proposed changes are intended to strengthen this function.

B. The changes do not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because the proposed changes do not affect any system, equipment, or plant operating procedure.

C. The changes do not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)) because no margin for safety is defined by the Administrative Controls Section of the Technical Specifications.

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

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

NRC Project Director: Lawrence Yandell, Acting Director

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

Date of amendment request: April 21, 1989

Description of amendment request: The proposed amendment would revise the Technical Specifications to remove existing requirements on the reactor coolant resistance temperature detector (RTD) bypass system, and replace them with requirements on fast-response thermowell-mounted RTDs. The proposed change reflects a design change, when approved by the staff, which will eliminate use of the RTD bypass system.

To support this request, the licensee submitted Westinghouse topical report WCAP-12058, "RTD Bypass Elimination Licensing Report for Beaver Valley Unit 1" which describes the extensive analyses, evaluation and testing performed to ensure the new design meets all safety and regulatory requirements. The changes to the Technical Specifications would reflect the characteristics (e.g., response time) of the fast-response RTDs.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists in accordance with 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 RTDs are not assumed to be precursors of accidents. However, their timely response has direct impacts on the consequences of accidents analyzed in the Final Safety Analysis Report (FSAR). The licensee stated that the new RTDs will have the same total response times as the existing RTDs with their associated manifold bypass system. Since total response times are not changed, and RTDs are not regarded as accident precursors, the answer to the first question is negative.

The proposed change would involve elimination of the bypass system, which is part of the reactor coolant boundary. This change will be performed in a manner consistent with the applicable standards, will preserve the existing design bases, and will not adversely affect the qualification of any other plant systems. The new RTDs are of a proven design currently used at other plants (e.g., Salem, Robinson). Therefore, no new accidents can be attributed to the new RTDs.

Finally, there is no change in design basis. The new design is expected to provide the same overall reliability, redundancy and diversity as the existing design. No accident assumptions will be relaxed or modified. Hence the answer to the last question is also negative.

The staff therefore proposes to determine that the requested amendment involves no significant hazards considerations.

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

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

NRC Project Director: John F. Stolz

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

Date of amendment request: April 21, 1989

Description of amendment request: The proposed amendments would revise the Technical Specifications of both units to delete Table 4.4.5, "Reactor Vessel Material Irradiation Surveillance Schedule" and associated surveillance requirement 4.4.9.1.c. The same table will be included in the Updated Final Safety Analysis Report (UFSAR) of each unit. Meanwhile, the bases section will also be revised to reference the UFSAR. The proposed changes will not alter any plant configuration or operational procedures since the program for surveillance of reactor vessel material will continue to be governed by 10 CFR Part 50 Appendix H. The current Table 4.4-5 is redundant to the regulation.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists in accordance with 10 CFR 50.92(c). A proposed amendment to an operating license for a facility involves no significant hazard 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.

As discussed above, there is no change to the plant configuration or operational procedures as a result of the proposed amendments. The proposed change is administrative. Thus the answers to questions (1) and (2) are negative. Furthermore, the design bases of the units are not altered and there is no relaxation of any safety margin. Thus the answer to question (3) is also negative.

The staff therefore proposes to determine that the requested amendments involve no significant hazards considerations.

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

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

NRC Project Director: John F. Stolz

Florida Power and Light Company, et al., Docket Nos. 50-335 and 50-389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment requests: April 4, 1989

Description of amendment requests: The proposed license amendments are intended to make corrections to typographical errors in the Administrative Controls section of the Technical Specifications, delete the specific composition list for the Company Nuclear Review Board (CNRB) and replace it with a general statement defining the requisite level of expertise for membership, and revise the Independent Safety Engineering Group (ISEG) reporting and administrative requirements for St. Lucie Unit 2.

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 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; 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 licensee provided the following discussion regarding the above three criteria.

Criterion 1

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

The proposed changes are administrative in nature and do not affect assumptions contained in the safety analyses nor do they affect Technical Specifications that preserve safety analysis assumptions. Additionally, these changes do not modify the physical design and/or operation of the plant. Therefore, the proposed changes do not affect the probability or consequences of accidents previously analyzed.

Criterion 2

Use of the modified specification would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The changes being proposed are administrative in nature and will not lead to material procedural changes or to physical modifications to the facility. Therefore, the proposed changes do not create the possibility of a new or different kind of accident.

Criterion 3

Use of the modified specification would not involve a significant reduction in a margin of safety.

The changes being proposed are administrative in nature and do not relate to or modify the safety margins defined in or required and maintained by the Technical Specifications.

The typographical corrections proposed do not affect any margin of safety. The deletion of the composition list of Company Nuclear Review Board (CNRB) membership and replacement with qualifications requirements guidelines will not decrease the effectiveness of this organization's independent review scope nor will there be a reduction in the collective talents of the CNRB.

The changes proposed to the Independent Safety Engineering Group (ISEG) administrative control and reporting requirements will focus the control, reports and reporting requirements of the ISEG to the Site Vice President - St. Lucie, Florida Power

& Light Company (FPL) and thus ensure the most efficient and effective use of the ISEG's products. However, changing the administrative control and reporting requirements will not affect any margin of safety.

Based on the above, [the Florida Power & Light Company] has determined that the proposed amendment does not (1) involve significant increase in the probability of consequences of an accident previously evaluated, (2) create the probability of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety; and therefore does not involve a significant hazard consideration.

The staff has reviewed the licensee's no significant hazards consideration determination and agrees with the licensee's analysis.

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

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 33450

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

NRC Project Director: Herbert N. Berkow

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 1, Appling County, Georgia

Date of amendment request: March 20, 1989

Description of amendment request: Hatch Unit 1 Technical Specification (TS) 4.6.F.2 currently requires reactor coolant conductivity sampling once every 4 hours when the continuous conductivity monitor is inoperable. The proposed change would revise TS 4.6.F.2 such that the sampling would be required only once every 24 hours when the reactor coolant temperature is less than or equal to 212° F. When coolant temperature is greater than 212° F, the sampling frequency would remain at once every 4 hours.

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 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; 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.

Basis for Proposed Change:

High conductivity of the reactor coolant may indicate the presence of chlorides in the coolant which can lead to stress corrosion cracking of the stainless steel components in contact with the coolant. The corrosion rate is temperature dependent. Normally, reactor coolant conductivity is monitored continuously by a conductivity monitor. During periods when the conductivity monitor is out of service, conductivity is measured by taking periodic samples of the reactor coolant. Since the corrosion rate is temperature dependent, the Standard Technical Specifications (STS) for boiling water reactors as well as the TS for Hatch Unit 2 recognize this fact by allowing a reduced sampling frequency of once every 24 hours when the coolant temperature is less than or equal to 212° F. At higher coolant temperatures, both the STS and the Hatch 2 TS require coolant conductivity sampling at 4-hour intervals at times when the continuous conductivity monitor is out of service. This proposed change would make the sampling requirements for Hatch Unit 1 equivalent to the requirements for Hatch Unit 2 and consistent with the requirements of the STS.

The licensee's March 20, 1989, submittal provided an evaluation of the proposed change with respect to the three standards, as follows:

1. This change does not involve a significant increase in the probability or consequences of an accident, because the operation of any plant equipment or system is not affected.

2. The possibility of a different kind of accident from any analyzed previously is not created by this change, since the change does not affect the operation of any plant equipment or system. Therefore, no new modes of plant operation are introduced, and no new accident types can result.

3. Margins of safety are not significantly reduced by this change, since the proposed change relaxes the surveillance interval only when the reactor coolant is less than or equal to 212° F at which temperature the corrosion rate is low. Additionally the change is consistent with the STS for reactor coolant sampling when the continuous monitor is inoperable. No other Specifications are affected by this change.

The staff has considered the proposed change and agrees with the licensee's evaluations with respect to the three standards.

On this basis, the Commission has determined that the requested amendment meets the three standards and, therefore, has made a proposed determination that the amendment application does not involve a significant hazards consideration.

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

Attorney for licensee: Bruce W. Churchill, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: David B. Matthews

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

Date of amendment request: March 17, 1989

Description of amendment request: The amendment would (1) modify the Technical Specifications (TS) for Unit 1 to make the definitions of Limiting Conditions for Operation and Surveillance Requirements consistent with the guidance provided in NRC Generic Letter 87-09, and would modify the Unit 2 TS to make the wording of Specifications 3.0.4, 4.0.3 and 4.0.4 consistent with the wording of Enclosure 4 to Generic Letter 87-09; and (2) the Bases for Unit 2 TS Sections 3.0.1 through 3.0.4 and Sections 4.0.1 through 4.0.5 would be replaced with revised Bases as provided in Enclosure 5 to Generic Letter 87-09.

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 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; 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 licensee's March 17, 1989, submittal provided an evaluation of the proposed changes with respect to these three standards.

Basis for Proposed Change 1:

This change will modify the wording of Unit 1 TS Definitions M and JJ, as well as Unit 2 TS Sections 3.0.4, 4.0.3, and 4.0.4 to be consistent with the guidance provided in Enclosure 1 to Generic Letter 87-09.

Proposed Change 1 does not involve a significant hazards consideration for the following reasons:

1. It does not involve a significant increase in the probability or consequences of an accident previously evaluated, because neither plant operation nor design is affected by the proposed change. The proposed change is administrative in nature and primarily serves to provide plant operating personnel with clearer guidance regarding compliance with LCOs and Action Requirements under all operating conditions.

2. It does not create the possibility of a new or different kind of accident from any previously evaluated, because no new modes of operation or design configuration are introduced. The proposed change serves to strengthen the existing TS requirements by eliminating some areas of confusion and interpretation, and providing a clear statement of the specification's intent.

3. It does not involve a reduction in the margin of safety, because the proposed change does not impact any numerical value in the Technical Specifications. The change serves to strengthen the philosophy of compliance with the Technical Specifications.

Basis for Proposed Change 2:

Proposed Change 2 will replace the entire Bases section 3/4.0 of the Unit 2 Technical Specifications with the 3/4.0 Bases provided in Enclosure 5 to Generic Letter 87-09.

Proposed Change 2 does not involve a significant hazards consideration for the following reasons:

1. It does not involve a significant increase in the probability or consequences of an accident previously evaluated, because the proposed change only serves to provide background information and explain the intent of Section 3/4.0. The proposed change does not in any way adversely affect the design, operation, or testing of the plant.

2. It does not create the possibility of a new or different kind of accident from any previously evaluated, because the proposed change is administrative in nature and does not introduce any new modes of operation or design configuration.

3. It does not involve a significant reduction in the margin of safety, because the proposed change provides explanatory information and does not impact any safety analysis.

The staff has considered the proposed changes and agrees with the license's evaluations with respect to the three standards.

On this basis, the Commission has determined that the requested amendments meet the three standards and, therefore, has made a proposed determination that the amendment application does not involve a significant hazards consideration.

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

Attorney for licensee: Bruce W. Churchill, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: David B. Matthews

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-424 and 50-425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: April 6, 1989

Description of amendment request: The proposed amendments would modify Technical Specification 4.5.2.h.1)b) to increase for Vogtle Unit 1 the maximum total pump flow rate for the centrifugal charging pump lines with a single pump running from 550 gpm to 555 gpm.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a 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.

In regard to the proposed amendment, the licensee has determined the following:

1. It has been determined that both system and component performance will not be adversely affected by the increase in flow. Therefore, the probability of previously analyzed accidents has not been increased. Additionally, since no new failure mode or new limiting single failure has been identified, the possibility of a different accident being created does not exist and the probability of a malfunction of safety related equipment has not been increased.

The increased CCP flow has been determined to have no impact or an insignificant effect on the safety analysis results. Therefore, the consequences of an accident previously evaluated in the FSAR [have] not been increased and the consequences of a malfunction of equipment [have] not become more severe. Therefore, the increase in the CCP flow from 550 gpm to 555 gpm does not result in any increase in radioactive releases as a result of normal operation or as a result of evaluated accidents.

As indicated in the above evaluations, the acceptance criteria for each of the safety analyses has not been exceeded. Therefore, there is no reduction in the margin of safety between the safety analysis assumptions and the Technical Specification values as defined in the basis to the Technical Specification.

The increase in flow will not affect the postulated causes of previously evaluated accidents. The minimum required flow has not changed, therefore the accidents evaluated with minimum flow assumptions are not affected by this change. The increase in maximum flow has been demonstrated, as discussed above, to be well below the maximum values assumed in the accident analyses. The potential increase in flow has been shown to have negligible [e]ffect on pump and motor reliability. Therefore, this revision to the maximum allowable pump flow with a single pump running from 550 gpm to 555 gpm for Unit 1 will not involve a significant increase in the probability [or] consequences of accidents previously evaluated.

2. This change in allowable maximum flow rate does not involve any physical change in the plant. Should future flow adjustments allow the pump to flow at 555 gpm, it will continue to operate within its designed capability and within the safety analyses assumptions. Therefore, this revision to the Technical Specification does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. As discussed above, the minimum flow requirements of Technical Specifications have not changed. Evaluations have been performed which conclude that the maximum flow assumption used in those analyses continue to envelope the allowable value in the revised Technical Specification. Therefore, the margin between the results of the analyses and the safety limit have not changed, and this revision to the Technical Specification does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's determination and concurs with its findings.

Accordingly, the Commission proposes to determine that the proposed change involves no significant hazards consideration.

Local Public Document Room location: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia 30830.

Attorney for licensee: Mr. Arthur H. Domby, Troutman, Sanders, Lockerman and Ashmore, Candler Building, Suite 1400, 127 Peachtree Street, N.E., Atlanta, Georgia 30043.

NRC Project Director: David B. Matthews

Gulf States Utilities Company, Docket No. 50-458, River Bend Station, Unit 1 West Feliciana Parish, Louisiana

Date of amendment request: August 29, 1986 as modified May 2, 1989.

Description of amendment request: The proposed amendment would add Technical Specifications for the Suppression Pool Pumpback System (SPPS). Limiting Condition for Operation (LCO), Action requirements, and Surveillance Requirements for the SPPS would be added to Technical Specification 3/4.5.3, Suppression Pool. Bases 3/4.5.3 would also be modified to add the SPPS. The May 2, 1989 submittal revised the proposed LCO by increasing the minimum subsystems required to be operable from one to two and including related Action Statements. In addition, the proposed Technical Specifications would include a statement that the provisions of Specification 3.0.4 do not apply. Specification 3.0.4 states:

3.0.4 Entry into an OPERATIONAL CONDITION or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

By specifying that Specification 3.0.4 is not applicable, entry into an Operational Condition would be allowed with one SPPS subsystem inoperable when the suppression pool is required. This would include startup.

The application for amendment to add Technical Specifications for the SPPS is to satisfy a November 18, 1985 commitment made by the licensee during the development of the Technical Specifications for the full power license. The NRC staff requested that Gulf States Utilities develop the Technical Specifications and propose a license amendment to implement them.

This notice supersedes the notice published in the October 22, 1986 **Federal Register** (51 FR 37512).

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 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 licensee provided an analysis that addressed the above three standards in the amendment application in the August 29, 1986 submittal.

The proposed change does not include a significant increase in the probability or consequences of an accident previously evaluated because the change only identifies the SPPS as a necessary subsystem to ensure operability of the suppression pool. This change does not involve a design change or physical change to the plant.

Thus, there is no increase in the probability or consequences of any accident previously evaluated.

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated because this change only provides explicit requirements to have the SPPS an identified as integral part of suppression pool system. This change does not involve a design change or physical change with respect to new or modified equipment, nor does it involve a change in the mode of operating existing equipment.

Thus, no new accident scenario is introduced by this clarification of requirements for suppression pool operability.

The proposed change does not involve a significant reduction in the margin of safety because this clarification of requirements for suppression pool operability significantly reduces the possibility of not considering SPPS as part of suppression pool operability, which would enhance safety rather than reduce the margin of safety.

The licensee provided additional analyses in the May 2, 1989 submittal:

The revision to the action requirements will not increase the probability or consequences of an accident previously evaluated or create the possibility of a new or different event because the system design and operation remains consistent with that provided in the Safety Analysis Report, therefore, plant response remains as originally evaluated.

The relief from the provisions of Specification 3.0.4 will not reduce the level of safety because one system is still required and the operability of the ECCS equipment is not effected by leakage in the crescent area. Because of the watertight ECCS cubicals, this evaluation has shown with one SPPS subsystem operable the plant response to a single failure will not result in a primary success path, as analyzed in the safety analysis report, being inhibited. The request to allow startup and changes in the operational condition with one subsystem operable also supports the basis of the Technical Specification.

The change will not reduce any identified margin of safety because the functional testing will increase the plant staff awareness of the systems ability to perform as described in the Safety Analysis Report. Because the pumps are used during normal plant operation, the knowledge of the loss of the remaining operable subsystem will be readily available.

In conclusion, the proposed operating change will not increase the possibility or the consequences of a previously evaluated event and will not create a new or different kind of accident from any previously evaluated. Also, the results of this request are within all acceptable criteria will respect to system components and design requirements. The ability to perform as described in the updated safety analysis report (USAR) is maintained and therefore, the proposed change does not involve a significant reduction in the margin of safety. Therefore, GSU proposes that no significant hazards are involved.

The staff has reviewed the licensee's no significant hazards consideration determination. Based on the review and the above discussions, the staff proposes to determine that the proposed changes do not involve a significant hazards consideration.

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

Attorney for licensee: Troy B. Conner, Jr., Esq., Conner and Wetterhahn, 1747 Pennsylvania Avenue, NW., Washington, DC 20006

NRC Project Director: Jose A. Calvo

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

Date of amendment request: August 19, 1988

Description of amendment request: The proposed amendment would revise the surveillance testing requirements for the feedwater and main steam line isolation valves and the main steam line isolation valve position switches. The proposed Technical Specification changes represent revisions to Section 4.2.7, Reactor Coolant System Isolation Valves, Table 4.6.2a, Instrumentation that Initiates Scram, and the Notes for Tables 3.6.2a and 4.6.2a.

Specifically, Surveillance Requirement 4.2.7.c is being proposed for revision to change the frequency of testing the feedwater and main steam line power-operated isolation valves from at least twice per week to at least once per quarter. The proposed once-per-quarter test frequency would reduce wear that is detrimental to seat tightness, and is in accordance with ASME Boiler and Pressure Vessel Code, Section XI, 1983 Edition with Summer 1983 Addendum, which is the edition of the ASME Code endorsed by 10 CFR 50.55a(g).

In addition, Surveillance Requirement 4.2.7.d would be added to incorporate the full closure test for the feedwater and main steam line isolation valves consistent with the requirements of ASME Boiler and Pressure Vessel Code Section XI, 1983 Edition with Summer 1983 Addendum. This test would be performed during each plant cold shutdown unless it has been performed in the previous three months (92 days).

The existing Surveillance Requirement 4.2.7.d would be renumbered to 4.2.7.e, a purely administrative revision.

The revision to Table 4.6.2a changes the frequency of the main steam line isolation valve position instrument channel test from once per three months to once per cold shutdown. This change is in accordance with recommendations provided by the Office of Nuclear Reactor Regulation. In its safety evaluation that accompanied a May 8, 1984 Memorandum for R. Starostecki from D. Eisenhut, Subject: Nine Mile Point 1 - Evaluation of Technical Specification Requirements for Main Steam Isolation Valve Limit Switch Testing, the Office of Nuclear Reactor Regulation recommended that the instrument channel test for these valves be conducted "prior to startup following plant shutdowns by actual closure of the main steam isolation valve(s), unless the test has been performed within the previous 92 days." The revisions to Table 4.6.2a incorporate this recommendation.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards (10 CFR 50.92(c)) for determining whether a significant hazards consideration exists. 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 licensee has provided the following analysis:

1) The revision to the test frequency of the feedwater valves and the main steam line isolation valves meets appropriate industry standards. The test frequencies are in accordance with ASME Boiler and Pressure Vessel Code Section XI, 1983 Edition with Summer 1983 Addendum. This edition has been approved by the Nuclear Regulatory Commission as indicated in 10 CFR 50.55a. Furthermore, the change in test frequency is consistent with the licensee's proposed Inservice Testing Program. The change in test frequency continues to provide the necessary number of tests to provide an indication of reliability while preventing unnecessary wear to the affected equipment. Therefore, no significant increase in the probability or consequences of an accident previously evaluated will occur.

The change in frequency for performing the main steam isolation valve limit switch testing is consistent with the above-cited safety evaluation performed by the Office of Nuclear Reactor Regulation. That evaluation indicates that the probability of the protection system failing to initiate the actuation of the equipment is and can be maintained acceptably low without testing the equipment during reactor operation. This change is requested to require performing the instrument channel test in the cold shutdown condition only. This test should be performed during plant shutdown in order to prevent an inadvertent reactor scram. As indicated in the above-cited safety evaluation, the function of the main steam isolation valve limit switches is to initiate a scram to terminate a main steam isolation valve closure transient. However, if the limit switches should fail, two other independent and diverse scram functions (reactor high pressure and high neutron flux) are available to terminate the transient, as noted in the Nine Mile Point Unit 1 FSAR Section XV.3.5, Main Steam Isolation Valve Closure with Scram. Therefore, the proposed change to the main steam line isolation valve limit switch testing will not significantly increase the probability or consequences of a main steam line accident.

2) The proposed change regarding the exercising of the main steam and feedwater isolation valves maintains the same type of testing practiced in the past; only the frequency has changed. The change affecting the testing of the main steam isolation valve limit switches is to require testing to be performed only during cold shutdown. Since there is no change in plant configuration to perform the tests, the possibility of a new or different kind of accident from any accident previously evaluated will not be introduced.

3) The change in test frequency continues to provide an accurate indication of reliability while preventing unnecessary wear on equipment. Therefore, a significant reduction in a margin of safety will not occur.

Based on the above, the staff proposes to determine that the proposed changes do not involve a significant hazards consideration.

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

Attorney for licensee: Troy B. Conner, Jr., Esquire, Conner & Wetterhahn, Suite 1050, 1747 Pennsylvania Avenue, NW., Washington, DC 20006.

NRC Project Director: Robert A. Capra

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

Date of amendment request: October 19, 1988

Description of amendment request: Technical Specifications 3.2.6 and 4.2.6, regarding the Inservice Inspection Program, would be revised to incorporate the requirements of NRC Generic Letter 88-01, which presents the staff positions concerning intergranular stress corrosion cracking (IGSCC) in austenitic stainless steel piping in boiling water reactors (BWRs). The technical bases for these staff positions are detailed in NUREG-0313, Revision 2, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping."

Basis for proposed no significant hazards consideration determination: The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. 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 licensee has provided the following analysis:

1. The proposed amendment incorporates the recommendations of NUREG-0313 Revision 2, "Technical Report on Material Selection and Process Guidelines for BWR Coolant Pressure Boundary Piping," as promulgated by Generic Letter 88-01. Niagara Mohawk has been complying with the requirements of NUREG-0313 Revision 1 since 1979. Since these inspection programs are not a ~~factor in~~ calculating accident probabilities or consequences, incorporating this later revision of NUREG-0313 has no affect on the probability or consequences of an accident previously evaluated.

2. The examinations required by the Inservice Inspection Program are normally performed during refueling and maintenance outages. These examinations are designed to detect service generated defects. Since these examinations do not affect the operation of plant equipment, no increase in the probability or consequences of an accident will result from the proposed changes.

3. The proposed changes incorporate the requirements of NUREG-0313 Revision 2 as promulgated by Generic Letter 88-01 for the inspection of austenitic BWR stainless steel piping. The new requirements imposed by Generic Letter 88-01 provide an increase in the level of safety by requiring augmented inspections of all austenitic materials. However, no credit is assumed in the calculation of the safety margin for inservice inspection. Therefore, there will be no reduction in the margin of safety.

Based upon the above, the staff proposes to determine that the proposed amendment involves no significant hazards consideration.

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

Attorney for licensee: Troy B. Conner, Jr., Esquire, Conner & Wetterhahn, Suite 1050, 1747 Pennsylvania Avenue, NW., Washington, DC 20006.

NRC Project Director: Robert A. Capra

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

Date of amendment request: April 25, 1989

Description of amendment request: The proposed amendment would revise Technical Specifications 3.1.7, 6.9.1 and the associated Bases for Sections 2.1.1 and 3.1.7 of Appendix A of the license to replace the values of cycle-specific parameter limits with a reference to the Unit 1 Core Operating Limits Report, which contains the values of those limits. In addition, the Core Operating Limits Report has been included in the Definitions Section of the Technical Specifications (TS) to note that it is the unit-specific document that provides these limits for the current operating reload cycle. Furthermore, the definition notes that the values of these cycle-specific parameter limits are to be determined in accordance with the Specification 6.9.1f. This Specification requires that the Core Operating Limits be determined for each reload cycle in accordance with the referenced NRC-approved methodology for these limits and consistent with the applicable limits of the safety analysis. Finally, this report and any mid-cycle revisions shall be provided to the NRC upon issuance. Generic Letter 88-16, dated October 4, 1988, from the NRC provided guidance to licensees on requests for removal of the values of cycle-specific parameter limits from TS. The licensee's proposed amendment is in response to this Generic Letter.

Basis for proposed no significant hazards consideration determination: The staff has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to 10 CFR 50.92(c), a proposed amendment to an operating license 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 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 revision to the License Condition is in accordance with the guidance provided in Generic Letter 88-16 for licensees requesting removal of the values of cycle-specific parameter limits from TS. The establishment of these limits in accordance with an NRC-approved methodology and the incorporation of these limits into the Core Operating Limits Report will ensure that proper steps have been taken to establish the values of these limits. Furthermore, the submittal of the Core Operating Limits Report will allow the staff to continue to trend the values of these limits without the need for prior staff approval of these limits and without introduction of an unreviewed safety question. The revised specifications with the removal of the values of cycle-specific parameter limits and that addition of the referenced report for these limits does not create the possibility of a new or different kind of accident for those previously evaluated. They also do not involve a significant reduction in the margin of safety since the change does not alter the methods used to establish these limits. Consequently, the proposed change on the removal of the values of cycle-specific limits does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Because the values of cycle-specific parameter limits will continue to be determined in accordance with an NRC-approved methodology and consistent with the applicable limits of the safety analysis, these changes are administrative in nature and do not impact the operation of the facility in a manner that involves significant hazards consideration.

The proposed amendment does not alter the requirement that the plant be operated within the limits for cycle-specific parameters nor the required remedial actions that must be taken when these limits are not met. While it is recognized that such requirements are essential to plant safety, the values of limits can be determined in accordance with NRC-approved methods without affecting nuclear safety. With the removal of the values of these limits from the Technical Specifications, they have been incorporated into the Core Operating Limits Report that is submitted to the Commission. Hence, appropriate measures exist to control the values of these limits. These changes are administrative in nature and do not impact the operation of the facility in a manner that involves significant hazards considerations.

Based on the preceding assessment, the staff believes this proposed amendment involves no significant hazards considerations.

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

Attorney for licensee: Troy B. Conner, Jr., Esquire, Conner & Wetterhahn, Suite 1050, 1747 Pennsylvania Avenue, NW., Washington, DC 20006.

NRC Project Director: Robert A. Capra

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

Date of amendment request: May 9, 1989

Description of amendment request: The proposed amendment would modify the Technical Specification (TS) as follows: (1) TS Table 3.3-6, "Radiation Monitoring for Plant Operation," would be changed to allow containment purge and exhaust isolation area monitors (RE41 and RE42) to be inoperable during performance of the containment integrated leak rate test (ILRT), (2) TS Table 3.3-11, "Fire Detection Instruments" would be changed to require that the fire protection instruments in the electrical penetration area (Elevation 24'6") be operable during the ILRT and (3) TS 3.7.12.2, "Spray and/or Sprinkler Systems" and TS Table 3.7-4, "Fire Hose Stations" would be changed to allow the inoperability of the containment cable penetration area sprinkler system and containment fire hose stations during the ILRT.

Basis for proposed no significant hazards consideration determination: Millstone Unit 3 TS 4.6.1.2 and Appendix J to 10 CFR Part 50 requires that Northeast Nuclear Energy Company (the licensee) perform a Type A, ILRT, for the primary containment at the specified test interval. While preparing to perform the ILRT during the Cycle 2/ Cycle 3 refueling outage, the licensee identified two areas where incompatibility exists between the requirements to perform the ILRT and other TS requirements to maintain certain components and systems operable during the ILRT. The following areas of inconsistency were identified by the licensee:

1. Radiation Monitoring - TS Table 3.3-6, Item 1a, requires that the containment area purge and exhaust isolation radiation detectors be maintained operable (in all modes). If the subject monitors become inoperable, the containment exhaust and purge valves must be maintained in the closed position per Action Statement 26.

The licensee has proposed that Action Statement 26 be revised to remove the requirements that the containment purge and exhaust isolation area radiation monitors (RE41 and RE42) be operable during the Type A containment ILRT.

During a Type A containment ILRT, the Millstone Unit No. 3 containment is pressurized to the calculated design basis accident containment pressure of 54.1 psia to verify containment leak tightness. The pressurization path is through the purge air supply piping, Containment Penetration Z86. The containment purge and exhaust system is interlocked with radiation monitoring instrumentation located inside containment. Since the radiation monitoring instrumentation is not designed to withstand a pressure of 54.1 psia, they will be removed from containment for the duration of the ILRT. Per Technical Specification 3.3.3.1, which references TS Table 3.3-6, the purge and exhaust valves must be isolated with less than minimum radiation monitoring instrumentation channels available. However, opening the purge air supply valve is required to conduct the ILRT and satisfy 10 CFR Part 50, Appendix J. Therefore, a revision to Action Statement 26 has been proposed to removed the requirement that the RE41 and RE42 radiation monitors be operable during the containment ILRT.

2. Fire Protection - TS 3.7.12.2, Item K and TS Table 3.7-4 requires the containment cable penetration area sprinkler system and the containment fire hose stations, to be operable, respectively. The licensee has indicated that the containment fire protection water system that enters containment at Penetration Z56 must be drained and vented to meet the provisions of the Millstone Unit No. 3 Final Safety Analysis Report (FSAR) Section 6.2.6 and the requirements of 10 CFR Part 50, Appendix J for performance of the ILRT.

Accordingly, the licensee has proposed that a footnote be added to Technical Specification Section 3.7.12.2 and TS Table 3.7-4 which exempts the containment cable penetration area sprinkler system and containment fire hose stations from operability requirements during Type A containment ILRT. To partially mitigate the proposed inoperability of the containment fire suppression systems, the licensee has proposed a footnote to Table 3.3-11 to include a requirement that fire detection instruments in the electrical penetration area, Elevation 24'6", be operable during the performance of Type A containment ILRT. All other fire detection instruments located within the containment area would not be required to be operable during the performance of a Type A containment ILRT. At the present time, TS Table 3.3-11 does not require the operability of any fire protection instrumentation, inside containment, during the ILRT.

Title 10, CFR 50.92, "Issuance of Amendment," contains standards for addressing the existence of no significant hazards considerations with regard to issuance of license amendments. The licensee has addressed the standards of 10 CFR 50.92, with regard to the proposed changes to the TS associated with the May 9, 1989 application, as follows:

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

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

The Type A ILRT is performed in Mode 5 with no personnel in containment. There are no design basis accidents which occur in Mode 5 and rely on either containment purge and exhaust radiation monitoring or the inside containment fire detection/suppression equipment. The only accidents which can occur in Mode 5 and require these functions are a loss of shutdown cooling and an inside containment fire.

Sufficient time exists following a loss of shutdown cooling for the operator to manually isolate the valves and prevent any releases from containment. Operator action is based on indications of a loss of shutdown cooling event. Thus, the change does not impact the consequences of a loss of shutdown cooling event.

During depressurization of the containment, grab samples will be obtained to verify that a radioactivity release is not occurring. Thus, it will limit the potential radiological consequences of the ILRT to an acceptable level.

The fire detection and suppression equipment is credited only in fire scenarios. The changes will permit the containment fire water isolation valves to be closed in order to measure containment leakage, but will require the fire detection instrumentation in the electrical penetration area to be operable. The operating fire detection components ensure that the operators will be alerted to a fire inside containment. As stated above, the plant procedure governing the Type A containment ILRT will require the cancellation of the ILRT and the opening of containment water isolation valves if both a smoke detection alarm is received and if any energized component/system operating within the containment trips simultaneously for any unknown reason during the test. Action statements within the containment leakage rate test procedure will allow the plant to take appropriate actions (open fire isolation valves) before any major fire damage occurs. Thus, the change does not impact the consequences of a postulated inside containment fire.

The containment purge and exhaust radiation monitoring equipment and containment fire detection/suppression system do not have the potential to initiate any previously analyzed accident. Operator action to isolate the purge and exhaust system or unisolate the containment fire water system, based on available indication, will negate the impact on the consequences of having these systems inoperable. For these reasons, the changes to the operability requirements of these systems do not increase the probability or consequence of any previously analyzed accident.

2. Create the possibility of a new or different kind of accident from any previously analyzed. The changes do not alter the way the plant is operated and only affects the containment ILRT. The change does not introduce new failure modes. For these reasons, the change does not have the potential to create a new type of accident from that previously analyzed.

3. Involve a significant reduction in a margin of safety. The changes do not impact any of the protective boundaries. The plant operators will be able to either isolate the containment purge and exhaust system or unisolate the containment fire water system (during the ILRT) based on available instrumentation. Thus, these safety functions will not be impacted by the change. The change does not increase the consequences of any design basis event. For these reasons, the change does not reduce the margin of safety.

The NRC staff has reviewed, and concurs in, the licensee's statement regarding "no significant hazards considerations" associated with the May 9, 1989 application for license amendment.

Accordingly, the staff has made a proposed determination that the application for amendment, dated May 9, 1989, involves no significant hazards consideration.

Local Public Document Room location: Waterford Public Library, 49 Rope Ferry Road, Waterford, Connecticut 06385.

Attorney for licensee: Gerald Garfield, Esquire, Day, Berry & Howard, One Constitution Plaza, Hartford, Connecticut 06103-3499.

NRC Project Director: John F. Stolz

Pacific Gas and Electric Company, Docket Nos. 50-275 and 50-323, Diablo Canyon Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Dates of amendment request: March 22, 1989 and May 15, 1989 (Reference LAR 89-03)

Description of amendment request: The proposed amendment would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant (DCPP) Unit Nos. 1 and 2 to

(1) Change TS 4.3.1.1, Table 4.3-1, Item 23, Seismic Trip, to increase the surveillance test interval (STI) for the seismic trip system actuating device operational test from 6 to 18 months to eliminate the need to perform seismic

trip system surveillance testing at power, and

(2) Change TS 3.3.1, Table 3.3-1, Item 23, Seismic Trip, to allow any one of the three seismic trip system channels to be bypassed for up to 72 hours for surveillance testing or maintenance while operating at power.

This request was previously noticed in the **Federal Register** on May 3, 1989 at 54 FR 18951. This replaces the previous notice.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a no significant hazards consideration exists as stated in 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 licensee in its submittal of March 22, 1989, evaluated the proposed changes against the significant hazards criteria of 10 CFR 50.92 and against the Commission guidance concerning application of this standard. Based on the evaluation given below, the licensee has concluded that the proposed changes do not involve a significant hazards consideration. The licensee's evaluation, as modified by the staff, is as follows:

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

Operation of the seismic trip system is not required or assumed to mitigate the consequences of any accident in the FSAR Update safety analyses. The seismic trip system component history demonstrates that component failures would not have prevented a reactor trip had a seismic event of the prescribed magnitude occurred. Because the system design does not permit reliable testing at power, two challenges to the reactor protection system have occurred during testing. Such challenges cause an increase in core damage frequency. Increasing the STI to allow testing to be performed during shutdown periods will eliminate the risk of inadvertent reactor trips and establishing an out of service time will allow for maintenance or component replacement at power.

Therefore, the proposed changes to increase the STI of the trip actuating device operational test to 18 months and...[allowing one of three channels to be bypassed for up to] 72 hours do not increase the probability or consequences of any accident previously evaluated.

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

There is no physical alteration to any plant system, nor is there a change in the method by which any safety related system performs its function. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes would potentially reduce the number of inadvertent reactor trips due to on-line surveillance testing and, therefore, would result in an increase in plant safety. Since the seismic reactor trip is not assumed to function for any of the Chapter 15 FSAR Update accident analyses, there is no affect on the margin of safety as defined in those analyses. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the proposed changes and the licensee's no significant hazards consideration determination and finds them acceptable. Therefore, the staff proposes to determine that these changes do not involve a significant hazards consideration.

Local Public Document Room location: California Polytechnic State University Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

Attorneys for licensee: Richard R. Locke, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120 and Bruce Norton, Esq., c/o Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Project Director: George W. Knighton

Pacific Gas and Electric Company, Docket Nos. 50-275 and 50-323, Diablo Canyon Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Dates of amendment request: May 12, 1989 (Reference LAR 89-05)

Description of amendment request: The proposed amendment would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant (DCPP) Unit Nos. 1 and 2 to change the diesel generator (DG) allowed outage time (AOT) to 7 days. Specific TS changes would include (1) revising the AOT requirement of TS 3.8.1.1 Action Statement b. to a 7-day AOT requirement for any one inoperable DG, and (2) revising the associated Bases to indicate the proposed AOT is an exception to the recommendations of Regulatory Guide 1.93.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a no significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license 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 licensee, in its submittal of May 12, 1989, evaluated the proposed changes against the significant hazards criteria of 10 CFR 50.92 and against the Commission guidance concerning application of this standard. Based on the evaluation given below, the licensee has concluded that the proposed changes do not involve a significant hazards consideration. The licensee's evaluation is as follows:

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

The Diablo Canyon offsite and onsite power systems are highly reliable. The 230kV and 500kV systems have been demonstrated to provide reliable offsite power sources for both units. The DCPP DG reliability history indicates that average reliability is higher than the requirements in Regulatory Guide 1.155, Station Blackout, and is higher than the industry average.

The risk and reliability evaluation determined that the probability of an accident previously evaluated does not significantly change by increasing the DG AOT from 72 hours to 7 days. The relative risk evaluation demonstrated that the relative risk remained low with an increased AOT from 72 hours to 7 days because of the improved maintenance possible with the 7-day AOT and the avoidance of multiple 72-hour AOTs.

Increasing the DG AOT does not involve physical alteration of any plant equipment and does not affect analysis assumptions regarding functioning of required equipment designed to mitigate the consequences of accidents. Further, the severity of postulated accidents and resulting radiological effluent releases will not be affected by the increased AOT.

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

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

Extending the DG AOT from 72 hours to 7 days does not necessitate physical alteration of the plant or changes in parameters governing normal plant operation.

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

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

As discussed above, the risk and reliability evaluations determined that the change in core melt frequency for a 7-day AOT compared with a 72-hour AOT is insignificant.

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

The NRC Staff has reviewed the proposed changes and the licensee's no significant hazards consideration determination and finds them acceptable. Therefore, the Staff proposes to determine that these changes do not involve a significant hazards consideration.

Local Public Document Room location: California Polytechnic State University Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

Attorneys for licensee: Richard R. Locke, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120 and Bruce Norton, Esq., c/o Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Project Director: George W. Knighton

Pacific Gas and Electric Company, Docket Nos. 50-275 and 50-323, Diablo Canyon Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Dates of amendment request: May 15, 1989 (Reference LAR 89-06)

Description of amendment request: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant (DCPP) Unit Nos. 1 and 2 to allow removal of the Boron Injection Tank from Units 1 and 2. The proposed BIT removal is consistent with the guidance provided in NRC Generic Letter 85-16, which concluded that there are inherent safety risks associated with the use of high concentrations of boron and that improved analysis methods are available to allow BIT removal. Specific TS changes would include: (1) Deletion of TS 3.5.4.1, "Boron Injection Tank", TS 3.4.4.2, "Heat Tracing", and the associated Bases, to allow for bypassing or removing the BIT and associated piping and components; (2) Revision of TS Table 3.3-5, "Engineered Safety Features Response Times", to make the safety injection response times consistent with BIT removal; and (3) Revision of TS Table 3.8-1 to change the function of the BIT inlet and outlet valves to charging injection valves.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a no significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment will not: (1) involved 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 or safety.

The licensee, in its submittal of May 15, 1989, evaluated the proposed change against the significant hazards criteria of 10 CFR 50.92 and against the Commission guidance concerning application of this standard. Based on the evaluation given below, the licensee has concluded that the proposed change does not involve a significant hazards consideration. The licensee's evaluation is as follows:

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

Analysis was performed for an "Accidental Depressurization of the Main Steam System" (FSAR Update Section 15.2.13) and "Major Secondary Steam System Pipe Rupture" (FSAR Update Section 15.4.2) with the BIT removed. For both cases after the reactor trip, the analysis determined that criticality is reattained due to plant cooldown, but the DNB design basis is met and no fuel failure will occur. Further analysis was performed to determine the impact of BIT removal on the containment mass and energy release and containment pressure and temperature response. It was shown that the containment pressure remained below its 47 psig design limit. The containment temperature response increased from the presently reported peak temperature value of 339 degrees F to 345 degrees F. PG&E has determined that the components inside containment critical to safety are not adversely affected by this small increase in temperature. Therefore, analysis results determined that the containment pressure transient response for the most limiting case assured pressure below design and the aggregate temperature response would not affect the current equipment qualification inside containment. Finally, analysis was performed assuming removal of the BIT to determine the mass and energy release due to steamline breaks outside containment assuming superheated steam release. Analysis results demonstrate that for the worst case main steamline break outside containment, all safety-related equipment required to mitigate the steamline break accident outside containment and structural components that would be both subject to the new superheat accident environment and necessary to mitigate the consequences of an accident would either function as designed or would be requalified or replaced.

The results of the safety injection response time evaluation demonstrated that delivery of borated water to the RCS meets all accident acceptance criteria.

The results of the above analyses demonstrate that consequences of previously evaluated events are not significantly increased. The results of the above analyses further demonstrate an increase in the probability of a return to criticality during a Condition II event (depressurization of the main steam system). However, there is no increase in the probability of fuel failure and releases remain within the guideline values of 10 CFR 20. Therefore, the equipment inside and outside containment necessary to mitigate the consequences of an accident would function as designed after modification and releases during depressurization of the main steam system remain within the guideline values of 10 CFR 20.

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

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

As discussed above, environmentally qualified equipment to provide emergency system functions inside and outside containment during a steamline break has been evaluated for the new environment that could result during accidents with the BIT removed. The analysis results demonstrated that this equipment will either still respond during accidents or will be requalified or replaced.

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

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

For both the "Accident Depressurization of the Main Steam System" (FSAR Update Section 15.2.13) and "Major Secondary Steam System Pipe Rupture" (FSAR Update Section 15.4.2), the Westinghouse analysis shows that the DNB design basis is met and no core damage results. Therefore, for the depressurization of the main steam system, release associated with this accident will remain within the guideline values set forth in 10 CFR 20 and for the major steam line break the radiation releases are within the guideline values set by 10 CFR 100. The safety injection response times continue to mitigate the consequences of LOCA and non-LOCA accidents with sufficient safety margin.

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

The NRC staff has reviewed the proposed changes and the licensee's no significant hazards consideration determination and finds them acceptable. Therefore, the staff proposes to determine that these amendments do not involve a significant hazards consideration.

Local Public Document Room location: California Polytechnic State University Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

Attorneys for licensee: Richard R. Locke, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120 and Bruce Norton, Esq., c/o Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Project Director: George W. Knighton

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

Date of amendment request: April 12, 1989

Description of amendment request: The licensee has provided the following description:

This application seeks to revise Appendix A of the Indian Point 3 Facility Operating License. Item 13 of Table 3.5-5 and Item 24 of Table 4.1-1 provide information regarding the temperature detection system in the Primary Auxiliary Building (PAB) of the Indian Point 3 Nuclear Power Plant. The proposed changes to the Technical Specifications revise these tables to reflect the sensor locations, and the operability and surveillance requirements of a new temperature detection system. Also included is the reorganization of the existing Auxiliary Boiler Feedwater Pump Building temperature sensors. The proposed change incorporates all temperature sensors into Item 13.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a 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 licensee has provided the following evaluation:

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

Response

The proposed license amendment reflects changes resulting from improvements to the temperature detection system in the PAB. Changes to the system were required as a result of the Steam Generator Blowdown System Upgrade and consequent high energy line break (HELB) analysis. The new temperature detection system serves the same function as the old system since it continues to provide for detection of line breaks in the piping penetration area. Improvements in the system include the provision of redundant detection instrumentation with a lower setpoint and shorter response time than that of the old system. These improvements do not involve an increase in the probability or consequences of an accident previously evaluated.

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

Response

The proposed license amendment reflects a change to the temperature detection system in the PAB. The change is necessary as a result of a new SGBD HELB analysis. The results of this analysis indicate the need for earlier rupture detection and automatic isolation of the Steam Generator Blowdown lines to prevent harsh environments in the PAB. The new temperature detection system satisfies these requirements by providing temperature sensors which annunciate at a lower setpoint and assist in the prevention of harsh environments by actuating closure of the blowdown isolation valves. These sensors are environmentally qualified and monitor the areas of the PAB where high energy lines are located. The sensors are not accident initiators. Hence, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

Response

The proposed license amendment reflects changes resulting from improvements to the temperature detection system which increase detection reliability and decrease response time. Hence, the new system does not involve a reduction in a margin of safety.

Based on the above, the NRC plans to determine that the proposed amendment 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 Project Director: Robert A. Capra

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

Date of amendment request: April 12, 1989

Description of amendment request: The licensee has provided the following description:

This application for amendment to the Indian Point 3 (IP3) Technical Specifications seeks to revise Paragraph 3.1.A.1.d of Appendix A regarding residual heat removal (RHR) pump operability during the cold shutdown condition with T_{avg} above 140° F. The change clarifies limiting conditions for operation of the Reactor Coolant System (RCS), ensuring consistency with existing specifications, and meeting the intent of Westinghouse Standard Technical Specifications (W STS).

Paragraphs 3.1.A.1.b through d provide Reactor Coolant Pump (RCP) and RHR Pump operating requirements during the conditions of hot and cold shutdown. Additionally, Paragraphs b and c allow for pump inoperability under stipulated conditions. The proposed change to paragraph 3.1.A.1.d would grant a similar provision. The proposed change would allow the operating RHR pump to be out-of-service for up to one (1) hour provided no operations are permitted that would cause dilution of the RCS boron concentration, and core outlet temperature is maintained at least 10° F below saturation temperature. The one hour allowed for the no pump running condition is not of sufficient duration to allow significant localized boron dilution due to stratification.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a 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 license made the following analysis of these changes:

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

Response:

The proposed change provides flexibility consistent with existing Technical Specifications and W STS, without compromising decay heat removal capability. Should the one operating pump become inoperable, a second pump is available for decay heat removal and Specification 3.3.A.7.a or b is applicable. Additionally, the one hour allowed for the no pump running condition is not of sufficient duration to allow significant localized boron dilution due to stratification. Combined with the requirement for no operations that could cause dilution, the probability of exceeding shutdown margin in any region of the core is not significantly increased. The requirement to maintain core exit temperature 10° F below saturation provides sufficient margin to the onset of boiling, including time to restore cooling before boiling occurs in any part of the core. Thus, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response:

The proposed change does not compromise the decay heat removal redundancy criteria set forth by the Commission's June 11, 1980 letter. In addition, changes to setpoints or hardware are not involved, and the operation of RCS/RHR temperature and flow instrumentation are not affected. Hence, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response:

In accordance with Specification 3.3.A.7, two RHR pumps are required to be operable during the cold shutdown condition above 140° F. Should the one operating pump become inoperable, a second pump is available for decay heat removal and Specification 3.3.A.7.a or b is applicable. Thus, the proposed change does not adversely affect existing specifications. In addition, the proposed change does not affect the operation of RCS/RHR surveillance instrumentation. Moreover, as discussed in response to question 1, the proposed change does not involve a significant increase in the probability of exceeding the shutdown margin in any region of the core. Since decay heat removal capability, system flow and temperature indication, and shutdown margin are not adversely affected, the proposed change does not involve a reduction in a margin of safety.

Based on the above, the staff proposes to determine that the proposed changes do 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 Project Director: Robert A. Capra

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: April 12, 1989

Description of amendment request: Take tritium sample directly from the spent fuel pool area rather than from the ventilation exhaust from that area.

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. In accordance with 10 CFR 50.92 the licensee has reviewed the proposed changes and has concluded as follows that they do not involve a significant hazards consideration:

Significant Hazards Consideration Evaluation

The proposed change to the Hope Creek Generating Station (HCGS) Technical Specifications:

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

The proposed amendment does not involve a change to any structure, component or system that affects the probability of any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). The proposed change will provide more accurate sampling results, thereby enhancing plant safety.

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

The proposed change in the sampling point for the measurement of tritium does not create the possibility for any accident. The revision merely provides for the use of a tritium sampling point that is more conservative than the one presently specified.

3. Does not involve a significant reduction in a margin of safety. Since the proposed sample location provides more accurate information regarding spent fuel pool area tritium levels than the presently specified grab sample point, the resulting increase in confidence in parameter measurement, hence detection capability, would enhance margins of safety.

The staff reviewed the licensee's determination that the proposed license amendment involves no significant hazards consideration and agrees with the licensee's analysis. Accordingly, the staff proposes to determine that the proposed license amendment does not involve a significant hazards consideration.

Local Public Document Room location: Pennsville Public library, 190 S. Broadway, Pennsville, New Jersey 08070

Attorney for licensee: Troy B. Conner, Jr., Esquire, Conner and Wetterhahn, 1747 Pennsylvania Avenue, NW., Washington, DC 20006

NRC Project Director: Walter R. Butler

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

Date of amendment request: July 15, 1988 and supplemented by letter dated April 25, 1989.

Description of amendment request: The proposed amendments make changes to the Administrative Controls, Section 6.0, of the Salem Generating Station Technical Specifications, Units 1 and 2. The first change involves the deletion of the offsite and onsite organization charts, Figures 6.2-1 and 6.2-2, and replaces them with more general requirements which capture the essential aspects of the organizational structure. Technical Specifications (TS) 6.1.1 and 6.1.2 have been supplemented with the necessary general requirements specified in Generic Letter 88-06, dated March 22, 1988.

The second change replaces the reference to the Vice President - Nuclear contained in TS 6.1.2 with the actual title, Vice President and Chief Nuclear Officer. For consistency, TS 6.2.1, 6.5.1.6, 6.5.1.8, 6.5.1.9, 6.5.2.4.2, 6.5.2.6, 6.5.2.7, 6.6.1, and 6.7.1, have also had the title, Vice President - Nuclear, replaced with the title, Vice President and Chief Nuclear Officer. This change is necessary because the title, Vice President - Nuclear Officer, and the Technical Specifications as currently structured should reflect this change. The Index is being revised to make it consistent with the aforementioned changes.

Basis for proposed no significant hazards consideration determination: The onsite and offsite organizations are currently defined by organization charts included in the Administrative Controls sections of the Salem Generating Station Technical Specifications (TS). As such, this requires that a License Amendment be processed for changes in organizational structure. The content requirements for the Administrative Controls section of the TS, which are specified in 10 CFR 50.36(c)(5) states that the TS contain the controls and provisions "...necessary to assure operation of the facility in a safe manner...," but does not specifically require the inclusion of detailed organization charts in the TS.

Since detailed organization charts are not specifically required by regulation, and since through experience the NRC staff has determined, "...that organization charts by themselves are of little help in ensuring that the administrative control requirements are met...," with appropriate changes to the administrative control requirements, the licensee proposes to remove the organization charts from the TS. The removal of organizational charts from the TS implements an improvement recommended by NRC in Generic Letter 88-06.

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; 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 licensee has analyzed the proposed amendment to determine if a significant hazards consideration exists:

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. The changes being proposed are administrative in nature and do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. For these reasons, the proposed changes do not affect the probability or consequences of accidents previously analyzed.

The NRC will continue to be informed of organizational changes through other required controls. The Code of Federal Regulations, Title 10, Part 50.34(b)(6)(i) requires that the applicants organizational structure be included in the Final Safety Analysis Report. Chapter 13 of the Salem Generating Station Updated Safety Analysis Report (UFSAR) contains a description of the organization with detailed organization charts, equivalent to or better than those which exist in the Technical Specifications. In accordance with 10 CFR 50.71(e), PSE&G submits annual updates to the UFSAR.

Changes to the organization described in the Quality Assurance (QA) Program are governed by 10 CFR 50, Appendix B, and 10 CFR 50.54(a)(3). Any changes to the organizational structure which have the potential to decrease the effectiveness of the QA Program require prior NRC approval. This amendment request proposes no changes to the current organizational structure, rather, it proposes to remove inaccurate information in favor of more general organizational requirements.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes being proposed are purely administrative in nature and will not lead to material procedure changes or to physical plant modifications. In addition, there are no management changes being proposed as a result of this amendment request. For these reasons, the proposed changes do not create the possibility of a new or different kind of accident.

3. The operation of the facility in accordance with the modified specification would not involve a significant reduction in a margin of safety. The changes being proposed are administrative in nature and do not relate to or modify safety margins defined in and maintained by the Technical Specifications (TS).

The changes proposed herein do not reduce the TS safety margin since all organizational responsibilities are being adequately implemented, all personnel are properly qualified, and controlling the organizational details in the UFSAR will be commensurate with controlling them in the TS.

Through PSE&G's strong Quality Assurance Program and our commitment to maintain only qualified personnel in positions of responsibility, it is assured that safety functions performed by the onsite and offsite organizations will continue to be performed at a high level of competence.

The staff has reviewed the licensees submittal and significant hazards analysis and concurs with the licensee's determination that the proposed amendment does not involve a significant hazards consideration. Therefore, the staff proposes to determine that the proposed amendment involves no significant hazards consideration.

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

Attorney for licensee: Mark J. Wetterhahn, Esquire, Conner and Wetterhahn, Suite 1050, 1747 Pennsylvania Avenue, NW., Washington, DC 20006

NRC Project Director: Walter R. Butler

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

Date of amendment request: March 23, 1989 and supplemented by letter dated April 14, 1989

Description of amendment request: The licensee proposes to modify the Salem Unit 2 Technical Specifications by deleting Technical Specification (TS) Table 3.8-1, "Containment Penetration Conductor Overcurrent Protective Devices," and to modify Bases 3/4.8.3 to require controls for maintaining the list of protective devices similar to those required for snubbers as described in Generic Letter 84-13, dated May 3, 1984. A specification for surveillances of fuses is being added to reflect the use of those fuses as overcurrent protective devices. Additionally, an identical specification would be added to the Unit 1 Technical Specifications, which currently has no specification for these devices, for consistency between units.

Basis for proposed no significant hazards consideration determination: Deleting Table 3.8-1 from the Technical Specifications and requiring administrative controls for the protective devices is similar to the requirements for snubbers as described in Generic Letter 84-13 and does not degrade compliance with TS 3.8.3.1. Technical Specification 3.8.3.1 will continue to require that the containment penetration conductor overcurrent protective devices be operable. The currently required surveillances will continue to be performed and the required corrective actions will be taken if the devices are found to be inoperable.

The list of containment penetration overcurrent protective devices and setpoints will be incorporated into a future revision to the updated final safety analysis report (UFSAR). Additionally, the setpoints for the subject devices will be incorporated into plant maintenance procedures and plant drawings which are controlled plant documents. Changes to the setpoints and devices are made through this controlled system in accordance with the licensee's quality assurance program and within the guidance of 10 CFR 50.59. Performing and documenting setpoint changes via the 50.59 process and including them in FSAR updates gives the staff adequate opportunity to review changes to the setpoint list.

Addition of an identical requirement to the Unit 1 Technical Specifications is being requested to achieve consistency between the Unit 1 and Unit 2 specifications.

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; 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 licensee has analyzed the proposed amendment to determine if a significant hazards consideration exists:

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

The proposed changes provide for the list of containment penetration conductor overcurrent protective devices to be maintained and controlled at the plant rather than in the Technical Specifications. The removal of the containment penetration conductor overcurrent protective device listing does not degrade the existing Technical Specification protective device operability and surveillance requirements nor does it affect the accident analysis. Therefore, this license amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not make any physical changes to the plant or changes in parameters governing normal plant operation. Therefore, the changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

As discussed above, the proposed changes do not degrade the existing protective devices' operability and surveillance requirements, nor do they effect the accident analysis. Therefore, changes do not involve a significant reduction in a margin of safety.

The staff has reviewed the licensee's submittal and significant hazards analysis and concurs with the licensee's determination that the proposed amendment does not involve a significant hazards consideration. Therefore, the staff proposes to determine that the proposed amendment involves no significant hazards consideration.

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

Attorney for licensee: Mark J. Wetterhahn, Esquire, Conner and Wetterhahn, Suite 1050, 1747 Pennsylvania Avenue, NW., Washington, DC 20006

NRC Project Director: Walter R. Butler

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

Date of amendment request: March 20, 1989

Description of amendment request: Proposed Change No. 200 is a request to revise Sections 3.14, "Fire Protection Systems Operability," and 4.15, "Fire Protection Systems Surveillance," of Appendix A Technical Specifications for the San Onofre Nuclear Generating Station, Unit 1 (SONGS 1). This change incorporates and supercedes Proposed Change No. 136 submitted by Amendment Application No. 120 dated June 8, 1984, as revised by SCE to NRC letter dated December 17, 1985, and Proposed Changes No. 159 and No. 162 submitted by Amendment Application No. 136 dated May 19, 1986.

This proposed change has resulted from modifications installed to comply with the safe shutdown requirements of 10 CFR Part 50 Appendix R and BTP 9.5.1, Appendix A.

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

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

RESPONSE: NO

Proposed Change No. 200 adds operability and surveillance requirements for equipment which has been installed to improve the Fire Protection Program at San Onofre Unit 1, in accordance with NRC requirements. This equipment will reduce the probability and/or consequences of a fire. Additionally, spurious operation of this equipment has been evaluated and determined not to significantly affect plant operation. Therefore, operation of San Onofre Unit 1 in accordance with Proposed Change 200 will not involve a significant increase in the probability or consequences of a previously evaluated accident.

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

RESPONSE: NO

Proposed Change No. 200 incorporates changes to the approved Fire Protection Program at San Onofre Unit 1 into the technical specifications. The proposed specifications assure that the required equipment is maintained operable or compensatory measures are implemented in compliance with the Fire Hazards Analysis for San Onofre Unit 1. Therefore, operation in accordance with Proposed Change No. 200 will not create the possibility of a new or different kind of accident from any previously evaluated.

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

RESPONSE: NO

Proposed Change No. 200 will assure that degradation of the required Fire Protection equipment will be detected and repaired or compensatory measures implemented. Therefore, operation of San Onofre Unit 1, in accordance with Proposed Change No. 200, will not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the analysis and, based on that review, it appears that the three criteria are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: George W. Knighton

System Energy Resources, Inc., et al., Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Dates of amendment request: December 19, 1988, as revised February 24, 1989.

Description of amendment request: The amendment would change the Technical Specifications (TS) by deleting TS 3/4.3.10, "Neutron Flux Monitoring Instrumentation," and modifying TS 3/4.4.1, "Recirculation System." Figure 3.4.1.1-1, "Power Flow Operating Map," would be changed to redefine flow stability regions. TS 3/4.4.1 would be changed to reflect the redefined regions of Figure 3.4.1.1-1. The Bases for TS 3/4.3.10 and TS 3/4.4.1 would be changed to reflect the changes in TS.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a no significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license 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 licensee has provided an analysis of no significant hazards considerations in its request for a license amendment. The licensee's analysis of the proposed amendment against the three standards in 10 CFR 50.92 is reproduced below.

1. These changes redefine the power/flow region and required operator actions from core stability considerations. While the changes allow for unrestricted operation in a region of the power/flow map where previously some monitoring requirements were applied, this region has been determined to be stable. In the power/flow area where instabilities are more likely to occur an immediate reactor shutdown is imposed. Revised operator actions are further defined to reduce the possibility of encountering instabilities and to rapidly terminate any instabilities should they occur in a region where instabilities are less likely to occur. The proposed changes have no affect on the core thermal-hydraulic instability phenomena. Therefore, they have no affect on its consequences, should an event occur. Therefore, the proposed changes

do not involve a significant increase in the probability or consequences of an instability event.

As these changes have no affect on the precursors to any accident previously evaluated, they will not affect the probability of an accident previously evaluated. Since previously evaluated events were evaluated for the entire power/flow map and allowed for the proposed operator actions, these changes have no affect on the consequences of those events previously evaluated. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. These changes do not involve any new design modifications or any new precursors to an accident. They only redefine areas of the operating map for core stability considerations and add conservative operator actions. Therefore, the proposed changes do not create the possibility of a new or different accident from those previously evaluated.

3. The proposed changes redefine regions of the power/flow map which are restricted from operation, eliminate the surveillance region, and redefine operator actions in the proposed regions.

Three stability regions are defined. Region A comprises the area above 100% rod-line and below 40% core flow. Region B comprises the area between the 80% and the 100% rod-lines and below 40% core flow. Regions A and B are restricted from operations. Region C comprises the operating areas above the 80% rod-line and between 40% and 45% core flow. Operation in Region C is allowed only for control rod withdrawals during startup.

Operator actions are revised consistent with the redefinition of Regions A, B and C. Upon entry into Region A, a single action is required, namely, an immediate reactor shutdown. Upon entry into Regions B or C, (unless operation in Region C is for control rod withdrawal during startup) an immediate action is required to exit the regions. This action can be either a reduction in thermal power or an increase in core flow. While operating in Regions B or C, the APRM neutron flux noise level will be observed. If a sustained APRM neutron flux noise level exceeding 10% peak-to-peak of rated thermal power is observed, an immediate reactor shutdown is required. In addition, with no reactor coolant system recirculation loops in operation in Regions A or B a single action is required, namely, an immediate reactor shutdown.

Region A comprises part of the restricted region in the current Technical Specifications (TS). The operator action proposed in this region, calling for immmediate reactor shutdown, render [sic] the proposed TS more conservative than current TS. Proposed operator actions in region B include immediate initiation of action to exit the region and an immediate reactor shutdown upon detection of oscillations. Thus, the proposed TS are more conservative for Region B as well. The proposed Region C comprises most of the current Detect and Suppress region and a triangular area where operation is currently restricted.

Although no instabilities have been observed in Region C, this region is maintained as a buffer zone. The proposed change restricts operation from this region except for certain startup activities. With the proposed actions, including an immediate reactor shutdown upon detection of oscillations, Region C is now more conservative than the current TS. The Detect & Suppress requirements for the triangular portion above the 45% flow line are deleted since the onset of instabilities in this area is unlikely and no restrictions are needed.

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

The licensee has concluded that the proposed amendment meets the three standards in 10 CFR 50.92 and, therefore, involves no significant hazards consideration.

The NRC staff has made a preliminary review of the licensee's no significant hazards consideration determination and agrees with the licensee's analysis. Accordingly, the Commission proposes to determine that the requested amendment does not involve a significant hazards consideration.

Local Public Document Room location: Hinds Junior College, McLendon Library, Raymond, Mississippi 39154.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Bishop, Liberman, Cook, Purcell and Reynolds, 1200 17th Street, NW., Washington, DC 20036

NRC Project Director: Elinor G. Adensam

System Energy Resources, Inc., et al., Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: April 18, 1989

Description of amendment request: The amendment would change the Technical Specifications (TS) by increasing the suppression pool low water level trip setpoint and allowable value in TS Table 3.3.8-2, "Plant Systems Actuation Instrumentation Setpoints." In addition, the description of the trip function and the suppression pool bottom reference elevation are revised to reflect the as-built plant conditions.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a no significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license 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 licensee has provided an analysis of significant hazards considerations in its request for a license amendment. The licensee has concluded, with appropriate bases, that the proposed amendment meets the three standards in 10 CFR 50.92 and, therefore, involves no significant hazards considerations.

The Commission has also provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on December 3, 1986 (51 FR 7744). The NRC staff has made a preliminary review of the licensee's submittal. A discussion of these examples as they relate to the proposed amendment follows.

One of the examples of actions involving no significant hazards consideration (i) involves an administrative change to correct an error or achieve consistency throughout the Technical Specifications. The changes to the description of the trip function and correction of the pool bottom reference elevation are similar to example (i). Another example of an action involving no significant hazards consideration (ii) is a change that constitutes an additional limitation, restriction, or control not presently included in the TS, e.g. a more stringent surveillance requirement. The increase in the suppression pool low water level setpoint is similar to this example because it is a more conservative setpoint. The function of this trip is to actuate the suppression pool makeup system following a loss-of-coolant accident (LOCA) when the water level in the suppression pool reaches the trip setpoint. Makeup is needed to assure that the drywell vents will have sufficient submergence to quench the steam from the blowdown during the LOCA. Increasing the low level trip setpoint will increase the minimum submergence of the vents and is, therefore, more conservative.

Based on the similarity of the changes to examples (i) and (ii) in 51 FR 7744, the NRC staff concludes that the changes in the proposed amendment are not likely to involve a significant hazards consideration. Accordingly, the Commission proposes to determine that the requested amendment does not involve a significant hazards consideration.

Local Public Document Room location: Hinds Junior College, McLendon Library, Raymond, Mississippi 39154.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Bishop, Liberman, Cook, Purcell and Reynolds, 1200 17th Street, NW., Washington, DC 20036

Tennessee Valley Authority, Docket No. 50-327, Sequoyah Nuclear Plant, Unit 1, Hamilton County, Tennessee

Date of amendment request: May 1, 1989 (TS 89-11)

Description of amendment requests: The Tennessee Valley Authority (TVA) proposes to revise the Sequoyah Unit 1 (SQN) Technical Specifications (TS). TVA is requesting a one-time extension of the 40 27 10-month, Type A, test interval in Surveillance Requirement (SR) 4.6.1.2.a. The proposed change would permit the third containment integrated leak rate test (ILRT) to be performed after February 1990, during the Unit 1 Cycle 4 refueling outage. TVA proposes to add a footnote to read as follows: "A one-time extension of the test interval is allowed for the third Type A test within the first 10-year service period provided unit shutdown occurs no later than May 1, 1990 and performace of the Type A testing occurs prior to unit restart following unit 1 cycle 4 refueling outage."

Basis for proposed no significant hazards consideration determination: TVA provided the following information in its application to support the proposed change to the SQN TS:

SQN's unit 1 entered its cycle 3 refueling outage on August 22, 1985. On December 15, 1985, unit 1 successfully completed its second periodic Type A test. Unit 1 returned to power operation on November 10, 1988, following an extended shutdown period. In accordance with the 40 2710-month test interval, SQN would be required to perform its third periodic unit 1 Type A test before February 15, 1990 (50 months). Application of the 40 2710-month test interval requires TVA to schedule a unit 1 shutdown sometime during [operating] cycle 4 for the sole purpose of performing a Type A test. TVA's current unit 1 cycle 4 schedule does not include a shutdown for the performance of a Type A test. The only outage currently scheduled during unit 1 cycle 4 is an 8-day outage for conducting an ice condenser flow passage inspection. This outage is scheduled to begin October 1, 1989. This inspection will be conducted while in mode 4 and will involve entering TS limiting condition for operation 3.6.5.1 for a 78-hour duration. The inclusion of a Type A test to this outage would require entry into mode 5. This would add an additional 2 to 3 days for temperature stabilization within containment; 14 days for setup, testing, and recovery; and 5 days for conducting mode 5 surveillance tests. The additional downtime described above would cost TVA approximately $2.5 million in replacement power costs. A forced outage for the sole purpose of performing a Type A test would similarly require a 22-day outage that would require 2-3 days of deadtime for temperature stabilization within containment. Based on the above cost options, TVA finds the extension of the Type A test to coincide with the unit 1 cycle 4 refueling outage to be economically prudent.

The proposed modification to the Type A test schedule is a temporary exemption to the required test interval. The proposed extension of the 40 2710-month test interval would enable unit 1 to complete its fourth fuel cycle without requiring a forced shutdown for [ILRT] test purposes. Considering that unit 1 has not experienced any unusual temperature or pressure excursions within the reactor containment building since the last Type A test and considering that no modifications have occurred that would have altered containment integrity, TVA finds no reason to suspect degradation in the unit 1 containment during the approximate 3-year shutdown period. It is important to note that the unit 1 containment structure was vented to the atmosphere during the extended outage. This configuration precluded any pressure oscillations that would be expected when the containment structure is in a closed condition (i.e., normal purging and/or venting). TVA investigated the option of performing a Type A test before the November 1988 unit 1 restart. This was discussed with NRC's Office of Special Projects. From these discussions, it was concluded that performance of two Type A tests within the 3-year extended outage would have imposed undue hardship with little or no compensating increase in the level of quality or safety. For these reasons, combined with the cost in man-hours for planning, scheduling, and conducting a Type A test, TVA requests a one-time exemption from the 40 2710-month test interval of unit 1 SR 4.6.1.2.a.

The Commission has provided Standards for determining whether a significant hazards determination exists as stated in 10 CFR 50.92(c). 10 CFR 50.91 requires that at the time a licensee requests an amendment, it must provide to the Commission its analyses, using the standards in Section 50.92, on the issue of no significant hazards consideration. Therefore, in accordance with 10 CFR 50.91 and 10 CFR 50.92, the licensee has performed and provided the following analysis:

TVA has evaluated the proposed TS change and determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of SQN in accordance with the proposed amendment will not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change is a one-time extension of the 40 2710-month, Type A, test interval as contained in SR 4.6.1.2.a. The purpose of the Type A test is to ensure that leakage through the primary containment and systems and components penetrating primary containment does not exceed allowable leakage rate values as specified in the TSs (SQN's limit is 0.75 La). Because the most likely leakage paths through containment are the penetrations, TVA completed a local leak rate test program on all penetrations and valves requiring Types B and C testing before unit 1 restart following the unit 1 cycle 3 refueling outage. This ensured that all Type B and C penetrations and valves were within the allowable containment leakage limit of 0.6 La. In addition, TVA performed [Surveillance Instruction] SI-254 to visually inspect the surfaces of the containment liner and the shield building for changes in appearance or other abnormal degradation before unit restart. Performance of these tests, coupled with the fact that unit 1 remained in cold shutdown condition during an extended 3-year period and did not experience any temperature excursions or pressure oscillations since the last Type A test, ensures that containment integrity was maintained during the 3-year shutdown period. On this basis, TVA has determined that the extension of the test interval would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any previously analyzed. No new accident scenarios are created by the proposed change because the one-time extension affects only the test frequency and does not affect the physical containment structure, the penetrations, or the facility. Previous Type A test results have shown that the leak rates for unit 1 have remained well below the 0.75-La limit. In addition, the unit 1 containment structure has not undergone modifications or been subjected to thermal or pressure excursions since the last Type A test that would have altered containment integrity. Because the 0.75-La leakage limit has not been compromised, the requested extension of the test interval will in no way create the possibility of a new or different kind of accident from any previously analyzed.

(3) Involve a significant reduction in a margin of safety. SQN's unit 1 was shut down for refueling in August 1985 and has remained in cold shutdown since that time [until its restart in November 1988]. The second regularly scheduled Type A test for unit 1 was successfully completed in December 1985. The data from the December 1985 test indicates a significant margin exists between the measured overall leak rate (0.05388 percent per day) and the 0.75-La limit (0.1875 percent per day). Because unit 1 has remained in cold shutdown and considering that no modifications have been performed on the containment boundary, the observed margin provided by the December 1985 test would not be expected to degrade beyond the 0.75-La leak rate limit. To ensure this margin is maintained, TVA completed a local leak rate program on all penetrations and valves requiring Types B and C testing before the November 1988 restart. In addition, TVA performed SI-254 to visually inspect the surfaces of the containment liner and the shield building for changes in appearance or other abnormal degradation. Based on these actions and the previous test margin, the one-time extension of the 40 2710-month, Type A, test interval would not involve a significant reduction in the margin of safety.

The staff has reviewed the licensee's no significant hazards consideration determination and agrees with the licensee's analysis. Therefore, the staff proposes to determine that the application for amendments involves no significant hazards considerations.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, E11 B33, Knoxville, Tennessee 37902.

NRC Assistant Director: Suzanne Black

Tennessee Valley Authority, Docket No. 50-327, Sequoyah Nuclear Plant, Unit 1, Hamilton County, Tennessee

Date of amendment request: May 5, 1989 (TS 89-14)

Description of amendment request: The Tennessee Valley Authority (TVA) proposes to revise the Sequoyah Unit 1 (SQN) Technical Specifications (TS). The proposed revision is to Surveillance Requirement (SR) 4.6.1.2.a, and the associated Bases Section 3/4.6.1.2, for primary containment integrity. The change is to delete the requirement that the third containment Type A (overall integrated containment leakage rate) test, of Appendix J to 10 CFR Part 50, be conducted during the shutdown for the 10-year plant inservice inspection.

Basis for proposed no significant hazards consideration determination: The requirement to conduct the third Type A test during the shutdown for the 10-year plant inservice inspection is in Appendix J to 10 CFR Part 50 and is a requirement on TVA independent of the SQN TS. TVA provided the following information in its application to support the proposed change:

SR 4.6.1.2.a requires that three Type A tests (containment integrated leak rate test (ILRT)) be conducted at 40 2710-month intervals during each 10-year service period with the third test to be conducted during shutdown for the 10-year plant inservice inspections (ISIs) (ISIs are required by 10 CFR 50.55.a). This (SR 4.6.1.2.a) TS implements the requirements of 10 CFR 50, Appendix J, Section IIID.1(a).

The third Type A test of the first 10-year service period for SQN unit 1 is presently scheduled to commence toward the end of the unit 1 cycle 4 refueling outage. This outage is scheduled to begin in April 1990. This ILRT schedule is contingent upon NRC approval of TVA's parallel ILRT extension request proposed under TS change 89-11 and its associated 10 CFR 50, Appendix J, exemption request (dated May 1, 1989). TVA intends to conduct the SQN unit 1, 10-year ISI during the unit 1 cycle 6 refueling outage that is currently scheduled to commence in March of 1993. TVA extended the SQN unit 1, 10-year ISI interval in accordance with the provisions of the American Society of Mechanical Engineers (ASME) [Code] Section XI, Article IWA-2400(c). The first SQN unit 1, 10-year ISI interval began July 1, 1981, and extends through September 15, 1994. Affirmation of this extension of the 10-year ISI is in preparation for submittal as part of TVA's commitment for providing NRC with SQN's revised ISI schedule.

SQN unit 1 entered its cycle 3 refueling outage on August 22, 1985. The second unit 1 ILRT was successfully completed on December 15, 1985. SQN unit 1 continued to remain in a cold shutdown condition (mode 5) over a 3-year period. Unit 1 returned to power operation on November 10, 1988. Because of this unusually long outage time, TVA submitted TS change 89-11 to request a one-time extension of the SQN unit 1 ILRT [test] frequency. NRC concurrence with the proposed TS change would allow the third unit 1 ILRT to be conducted during the unit 1 cycle 4 refueling outage. This one-time extension would require unit 1 shutdown for refueling no later than May 1, 1990. The 3-year, unit 1 shutdown period also resulted in adjustments to the unit 1, 10-year ISI interval in accordance with the provisions of ASME [Code] Section XI, Article IWA-2400(c). The proposed ILRT [test] extension and the adjustment in the 10-year ISI interval imposed separate timeframes for the required performance of the unit 1 ILRT and the scheduled 10-year ISI. To account for this separation, TVA is submitting the [above] request [application dated May 5, 1989] that would allow the third unit 1 ILRT and the 10-year ISI to be uncoupled and performed in separate refueling outages.

The Commission has provided Standards for determining whether a significant hazards determination exists as stated in 10 CFR 50.92(c). 10 CFR 50.91 requires that at the time a licensee requests an amendment, it must provide to the Commission its analyses, using the standards in Section 50.92, on the issue of no significant hazards consideration. Therefore, in accordance with 10 CFR 50.91 and 10 CFR 50.92, the licensee has provided the following analysis:

TVA has evaluated the proposed TS change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of SQN in accordance with the proposed amendment will not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The uncoupling of the third Type A test schedule from the 10-year ISI schedule does not involve a change in the test/inspection methodology or acceptance criteria from those previously (and currently) analyzed in the SQN Final Safety Analysis Report. The proposed change does not involve a change to the facility or modifications to equipment/ components or hardware; therefore, the probability or consequences of an accident previously evaluated have not increased.

(2) Create the possibility of a new or different kind of accident from any previously analyzed. The proposed change would allow separate timeframes for the required performance of the third Type A test and the scheduled 10-year ISI. This separation does not introduce any new type of accident or malfunction since the surveillance test frequency, acceptance criteria, and test/inspection methods remain unchanged. Conducting the third Type A test in a separate outage from the 10-year ISI will not result in any design or hardware changes and therefore does not create the possibility for a new or different kind of accident from any previously analyzed.

(3) Involve a significant reduction in a margin of safety. The proposed change will not reduce the margin of safety as defined in the basis of SQN TS. The Bases for TS 3/4 6.1.2, "Containment Leakage," states, "The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR 50." Compliance with the 10 CFR 50, Appendix J, requirements would continue to be maintained with the single exception that allows the third Type A test and the 10-year ISI not to be performed during a common unit outage. This uncoupling causes no reduction in the margin of safety since no changes were made to the containment test frequency or the containment leakage limits assumed in the accident analysis.

The staff has reviewed the licensee's no significant hazards consideration determination and agrees with the licensee's analysis. Conducting the 10-year ISI at Unit 1 during an outage later than the Cycle 4 refueling outage is in accordance with the ASME Code. Therefore the staff proposes to determine that the application for amendments involves no significant hazards considerations.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, E11 B33, Knoxville, Tennessee 37902.

NRC Assistant Director: Suzanne Black

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

Date of amendment request: April 28, 1989.

Description of amendment request: The amendment would primarily reflect organizational changes, correct typographical errors, correct inconsistencies, and clarify the intent of certain technical specifications (TS). One proposed change would remove a reference in TS to ANSI N18.7-1976 and replace it with a reference to the Operational Quality Assurance Program (OQAP) Description, which makes the same commitment as the existing TS. Another change would revise the basis for a TS to reflect a design modification conducted under 10 CFR 50.59. The proposed changes would not decrease the effectiveness of the TS.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. 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 licensee has evaluated the proposed organization changes against the standards provided above and has determined that the changes will not decrease the effectiveness of the WPSC nuclear organization and would involve no significant hazards consideration. The staff agrees with this evaluation. The licensee has also evaluated the proposed change that would reference the OQAP Description, rather that the individual sections of ANSI N18.7-1976, and has determined that there are no significant hazards associated with this change. The staff agrees with this evaluation. The staff also agrees with the licensee's determination that there are no significant hazards associated with a revision to the TS basis to reflect a design modification.

The Commission has provided examples (51 FR 7751) of amendments which are not likely to involve no significant hazards considerations. One of these examples, (i), states: "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." The other proposed changes are similar to this example, and are, therefore, not likely to involve significant hazards considerations.

Based on the above, the staff proposes to determine that the proposed amendment would involve no significant hazards considerations.

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

Attorney for licensee: David Baker, Esq. Foley and Lardner, P. O. Box 2193 Orlando, Florida 31082.

NRC Project Director: John N. Hannon.

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 either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

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

Connecticut Yankee Atomic Power Company, Docket No. 50-213, Haddam Neck Plant, Middlesex County, Connecticut

Date of application amendment: March 31, 1989

Brief Description of amendment request: The proposed amendment would revise the one-time relaxation of the containment integrity technical specifications issued as License Amendment No. 112, to allow the four containment air recirculation fan motor heat exchangers to be cleaned or replaced while at power.

Date of Individual Notice in **Federal Register:** May 12, 1989 (54 FR 20659).

Expiration date of individual notice: June 12, 1989

Local Public Document Room location: Russel Library, 123 Broad Street, Middletown, Connecticut 06457.

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

Date of amendment request: March 30, 1989

Brief Description of amendment: The proposed license amendment would modify the Technical Specifications 3/4.6.5.1 Secondary Containment and Definition 1.36 to reflect design modifications to the reactor building's railroad bay air lock doors.

Date of publication of individual notice in **Federal Register:** April 28, 1989 (54 FR 18372).

Expiration date of individual notice: May 30, 1989

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

Indiana Michigan Power Company, Docket No. 50-315, Donald C. Cook Nuclear Plant, Unit No. 1, Berrien County, Michigan

Date of application for amendments: October 14, 1988 as supplemented December 30, 1988

Brief description of amendment: This amendment revises the TSs to allow operation of future reload cycles of D. C. Cook, Unit 1 at reduced primary coolant system temperature and pressure conditions. The reduced temperature and pressure (RTP) conditions will decrease the steam generator U-tube stress corrosion cracking of the type observed at D. C. Cook, Unit 2.

Date of publication of individual notice in **Federal** Register: April 19, 1989 (54 FR 15851).

Expiration date of individual notice: May 5, 1989

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

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

Date of amendment request: June 10, 1985, as supplemented December 6, 1985 and May 16, July 14, July 28, and November 18, 1988 and April 5, 1989.

Description of amendment request: The proposed amendment to Virgil C. Summer Nuclear Station Technical Specification (TS) would reduce the number and severity of starts of the emergency diesel generators, thereby decreasing engine wear and increasing reliability. This proposed change was originally noticed as a proposed no significant hazards consideration on July 17, 1985 at 50 FR 29016 and renoticed on June 4, 1986 at 50 FR 20373, and August 24, 1988 at 53 FR 32295.

Date of publication of individual notice in **Federal Register:** April 26, 1989 (54 FR 18055).

Expiration date of individual notice: May 26, 1989.

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

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

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

Notice of Consideration of Issuance of Amendment to Facility Operating License 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, the Gelman Building, 2120 L Street, NW., Washington, DC, 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, DC 20555, Attention: Director, Division of Reactor Projects.

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

Date of amendment request: April 24, 1989 as supplemented on May 5, 1989

Brief description of amendment: The amendment changes ANO-1 license condition 2.c.(1) to increase the authorized steady state reactor core power level to a maximum of 2054 megawatts thermal, which is 80% of full power (2568 megawatts thermal).

Date of issuance: May 16, 1989

Effective date: May 16, 1989

Amendment No.: 120

Facility Operating License No. DPR-51. Amendment revised the license.

Date of initial notice in **Federal Register:** April 28, 1989 (54 FR 18366). The May 5, 1989 submittal provided additional clarifying information and did not change the proposed finding of the initial notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 16, 1989.

No significant hazards consideration comments received: No.

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

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

Date of applications for amendment: December 12, 1986 as supplemented on April 27, 1989.

Brief description of amendment: This amendment changes the Arkansas Nuclear One, Unit 2 Technical Specifications to make several editorial, clarifying, and administrative corrections. The changes removed typographical errors, revised wording to cite appropriate references, and provided consistent terminology.

Date of issuance: May 16, 1989

Effective date: May 16, 1989

Amendment No.: 94

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

Date of initial notice in **Federal Register:** May 20, 1987 (52 FR 18972). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 16, 1989.

No significant hazards consideration comments received: No.

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

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: December 29, 1988

Brief description of amendment: Revises the operability requirements for the service water system to require three operable service water pumps on the essential header and two operable service water pumps on the non-essential header whenever the reactor is above 350° F. Also adds a requirement to maintain isolation between the two headers.

Date of issuance: May 8, 1989

Effective date: May 8, 1989

Amendment No.: 139

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

Date of initial notice in **Federal Register:** March 22, 1989 (54 FR 11826). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 8, 1989.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: December 19, 1988

Brief description of amendment: This amendment revises the Big Rock Point Plant Facility Operating License to allow for an increase in the amount of by-product material the plant may possess and use as sealed sources from 10.5 curies of Cesium-137 to 45 curies of Cesium-137.

Date of issuance: May 2, 1989

Effective date: May 2, 1989

Amendment No.: 96

Facility Operating License No. DPR-6. The amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** February 22, 1989 (54 FR 7630). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 2, 1989.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 30, 1985, supplemented January 13, 1986

Brief description of amendment: This amendment revises the Technical Specifications by deleting the operability requirements for the high pressure safety injection (HPSI) flow instruments in Table 3.17.4.

Date of issuance: May 12, 1989

Effective date: May 12, 1989
Amendment No.: 121
Provisional Operating License No. DPR-20. The amendment revises the Technical Specifications.
Date of initial notice in Federal Register: August 28, 1985 (50 FR 34938). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 12, 1989.
No significant hazards consideration comments received: No.
Local Public Document Room location: Van Zoeren Library, Hope College, Holland, Michigan 49423.

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

Date of application for amendments: June 19, 1987, as supplemented March 10, 1989.
Brief description of amendments: The amendments modified the Technical Specifications to add changes required by NRC Generic Letter 85-09, "Technical Specifications for Generic Letter 83-28, Item 4.3," related to automatic actuation of the shunt trip attachment on reactor trip breakers.
Date of issuance: May 9, 1989
Effective date: May 9, 1989
Amendment Nos.: 63 and 57
Facility Operating License Nos. NPF-35 and NPF-52. Amendments revised the Technical Specifications.
Date of initial notice in Federal Register: April 5, 1989 (54 FR 13763). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 9, 1989.
No significant hazards consideration comments received: No.
Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

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

Date of application for amendments: June 19, 1987, as supplemented February 24 and November 23, 1988, and January 6, 1989.
Brief description of amendments: The amendments modified the Technical Specifications to authorize a one-time extension of the allowed outage times for the control area ventilation system to provide for system modification.
Date of issuance: May 12, 1989
Effective date: May 12, 1989
Amendment Nos.: 95 and 77
Facility Operating License Nos. NPF-9 and NPF-17: Amendments revised the Technical Specifications.
Date of initial notice in Federal Register: February 22, 1989 (54 FR 7632). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 12, 1989.
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: January 18, 1989
Brief description of amendment: The amendment revises Table 3.6-1, "Containment Penetrations," to identify penetrations 57-3 and 57-4 as spares. The licensee plans to cut the tubing associated with these penetrations and cap the tubing ends.
Date of issuance: May 15, 1989
Effective date: May 15, 1989
Amendment No.: 140
Facility Operating License No. DPR-66. Amendment revised the Technical Specifications.
Date of initial notice in Federal Register: February 22, 1989 (54 FR 7633). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 1989.
No significant hazards consideration comments received: No
Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001.

Florida Power Corporation, et al., Docket No. 50-302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of application for amendment: April 14, 1983, as modified December 13, 1983, July 25, 1984, January 24 and October 16, 1986 and August 18, 1987.
Brief description of amendment: This amendment clarifies certain administrative controls, and modifies the audit frequencies of the Security, Emergency, and Fire Protection plans to be consistent with 10 CFR 73.40(d). The requested change to Figures 6.2-1 and 6.2-2 was denied.
Date of issuance: May 5, 1989
Effective date: May 5, 1989
Amendment No.: 111
Facility Operating License No. DPR-72: Amendment revised the Technical Specifications.
Date of initial notice in Federal Register: February 24, 1984 (49 FR 7035). The letters dated July 25, 1984, January 24 and October 16, 1986 and August 18, 1987 provided supplemental information which did not alter the staff's initial determination of no significant hazards consideration. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 5, 1989.
No significant hazards consideration comments received: No.
Local Public Document Room Location: Crystal River Public Library, 668 N.W. First Avenue, Crystal River, Florida 32629

Florida Power Corporation, et al., Docket No. 50-302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of application for amendment: March 31, 1983, as supplemented June 22, 1983 and revised February 24, May 31, and December 31, 1984
Brief description of amendment: This amendment adds requirements to the Technical Specifications for the reactor coolant system high point vents.
Date of issuance: May 8, 1989
Effective date: May 8, 1989
Amendment No.: 112
Facility Operating License No. DPR-72. Amendment revised the Technical Specifications.
Date of initial notices in Federal Register: December 21, 1983 (48 FR 54504). and April 5, 1989 (54 FR 13764). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 8, 1989.
No significant hazards consideration comments received: No.
Local Public Document Room Location: Crystal River Public Library, 668 N.W. First Avenue, Crystal River, Florida 32629

Florida Power Corporation, et al., Docket No. 50-302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of application for amendment: June 22, 1983, as revised February 24, 1984.
Brief description of amendment: This amendment revises the TS by replacing the requirement that a hydrogen analyzer and a gas chromatograph be operable with a requirement that two hydrogen monitors be operable. It also establishes surveillance requirements and gives actions to be taken should one or both hydrogen monitors be found inoperable.
Date of issuance: May 8, 1989
Effective date: May 8, 1989
Amendment No.: 113
Facility Operating License No. DPR-72. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 5, 1989 (54 FR 13764). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 8, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room Location: Crystal River Public Library, 668 N.W. First Avenue, Crystal River, Florida 32629

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

Date of application for amendment: January 25, 1985

Brief description of amendment: This amendment deletes license conditions 2.C.10 and 2.C.11 from the St. Lucie Plant, Unit 2 Operating License No. NPF-16.

Date of Issuance: May 17, 1989

Effective Date: May 17, 1989

Amendment No.: 41

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

Date of initial notice in **Federal Register:** May 21, 1985 (50 FR 20976). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 17, 1989.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: December 4, 1987

Brief description of amendment: The amendment modifies Appendix A Technical Specifications by revising the Specifications related to Fire Protection systems at TMI-2.

Date of Issuance: May 15, 1989

Effective date: May 15, 1989

Amendment No.: 34

Facility Operating License No. DPR-73. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 8, 1989 (54 FR 6194). The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated May 15, 1989.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: April 19, 1988 as supplemented April 19, 1989

Brief description of amendment: This amendment changes the Technical Specifications to add Limiting Conditions for Operation and Surveillance Requirements for Containment Vent and Purge Valves and the Standby Gas Treatment System.

Date of issuance: May 15, 1989

Effective date: May 15, 1989

Amendment No.: 129

Facility Operating License No. DPR-46. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 18, 1988 (53 FR 17790). The April 19, 1989 submittal provided additional specifications that clarified the initial submittal and did not change the proposed finding of the initial notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 1989.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: January 26, 1989

Brief description of amendment: The Technical Specification change deletes the requirement to verify uniformity of air flow distribution across the charcoal absorber banks and HEPA filters of the Standby Gas Treatment System once per operating cycle.

Date of issuance: May 12, 1989

Effective date: May 12, 1989

Amendment No.: 32

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

Date of initial notice in **Federal Register:** April 5, 1989 (54 FR 13767). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 12, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: Waterford Public Library, 49 Rope Ferry Road, Waterford, Connecticut 06385.

Northern States Power Company, Docket No. 50-263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of application for amendment: May 1, 1985 as supplemented November 22, 1985 and November 3, 1986.

Brief description of amendment: This amendment revises the Technical Specification to: (1) restrict purge and vent valve operations above cold shutdown to the 2-inch by pass flow path except for inerting and deinerting containment; (2) require containment purge and vent valve seal seat maintenance at five year intervals; (3) specify the maximum operating time for containment purge and vent valve operation to 15 seconds; (4) reduce the number of outboard valves for "drywell purge inlet" to one and add a new table entry "suppression chamber purge inlet" with one outboard valve; (5) change the normal position specified for all drywell and suppression chamber vent and purge to "closed;" and (6) make other editorial changes related to these changes.

Date of issuance: May 10, 1989

Effective date: May 10, 1989

Amendment No.: 64

Facility Operating License No. DPR-22. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** October 23, 1985 (50 FR 43031). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 10, 1989.

No significant hazards consideration comments received: No.

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

Pacific Gas and Electric Company, Docket Nos. 50-275 and 50-323, Diablo Canyon Nuclear Power Plant, Units 1 and 2, San Luis Obispo County, California

Date of application for amendments: November 29, 1988, as supplemented by letters dated December 9, 1988 and February 17, 1989 (Reference LAR 88-08)

Brief description of amendments: The amendments revised the Technical Specifications to allow the use of Westinghouse VANTAGE 5 fuel assemblies in the reactors.

Date of issuance: May 10, 1989.

Effective date: Upon completion of Cycle 3 for Unit 1.

Amendment Nos.: 37 and 36

Facility Operating License Nos. DPR-80 and DPR-82: Amendments changed the Technical Specifications.

Date of initial notice in **Federal Register:** February 22, 1989 (54 FR 7639). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 10, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: California Polytechnic State University Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

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

Date of application for amendments: November 9, 1988

Brief description of amendments: Removal of drywell floor drain sump flowrate monitors from the Technical Specifications.

Date of issuance: May 4, 1989

Effective date: May 4, 1989

Amendment Nos.: 87 and 53

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

Date of initial notice in **Federal Register:** December 14, 1988 (53 FR 50334). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 4, 1989.

No significant hazards consideration comments received: No

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

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

Date of application for amendments: October 14, 1988

Brief description of amendments: Deleted erroneous Control Room Emergency Outside Air Supply System (CREOASS) surveillance requirements for Radiation and Reactor Building Isolation signals for Technical Specification 4.7.2.d.2.

Date of issuance: May 5, 1989

Effective date: May 5, 1989

Amendment Nos.: 88 and 54

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

Date of initial notice in **Federal Register:** December 14, 1988 (53 FR 50333). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 5, 1989.

No significant hazards consideration comments received: No

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

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

Date of application for amendments: December 12, 1988

Brief description of amendments: The amendments changed the Technical Specifications by revising the load profiles for 125 v dc battery banks.

Date of issuance: May 10, 1989

Effective date: May 10, 1989

Amendment Nos.: 89 and 55

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

Date of initial notice in **Federal Register:** April 5, 1989 (54 FR 13768). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 10, 1989.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: February 2, 1989

Brief description of amendment: Technical Specification changes to support Susquehanna Steam Electric Station Unit 1 Cycle 5 operations with Advanced Nuclear Fuels Corporation 9X9 reload fuel.

Date of issuance: May 15, 1989

Effective date: As of the date of issuance to be implemented upon startup for Cycle 5 operations currently scheduled for June 2, 1989.

Amendment No.: 90

Facility Operating License No. NPF-14: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 5, 1989 (54 FR 13767). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 1989.

No significant hazards consideration comments received: No

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

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

Date of application for amendments: July 31, 1979 as amended on June 4, 1984 and September 15, 1986.

Brief description of amendments: These amendments revised the Technical Specifications to incorporate a 90-hour purging restriction, definitions of conditions requiring no justification for purging, limitations on the use of the Standby Gas Treatment System (SGTS), operability requirements for the SGTS, additional TS for the containment purge and vent isolation valves and to correct certain valve and penetration numbers.

Date of issuance: May 8, 1989

Effective date: Units 2 and 3; effective within 60 days of the date of issuance except that the inflatable seal program specified in Technical Specification 4.7.E.1 shall become effective during the first refueling outage commencing six months following issuance of these amendments.

Amendments Nos.: 144 and 146

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

Date of initial notice in **Federal Register:** November 19, 1986 (51 FR 41864). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 8, 1989.

No significant hazards consideration comments received: No

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126.

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

Date of application for amendments: March 10, 1989 as supplemented May 5, 1989.

Brief description of amendments: These amendments revised the frequency for calibration of the Source Range Monitor and Intermediate Range Monitor detector not in startup position instrumentation.

Date of issuance: May 12, 1989

Effective date: May 12, 1989

Amendments Nos.: 145 and 147

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

Date of initial notice in **Federal Register**: March 21, 1989 (54 FR 11599). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 12, 1989.

No significant hazards consideration comments received: No

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126.

Power Authority of the State of New York, Docket No. 50-33, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of application for amendment: June 10, 1988

Brief description of amendment: The amendment clarifies the sections dealing with the Crescent Area Ventilation and associated Limiting Condition for Operation and Surveillance Testing inconsistencies.

Date of issuance: May 4, 1989

Effective date: May 4, 1989

Amendment No.: 126

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

Date of initial notice in **Federal Register**: March 8, 1989 (54 FR 9926). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 4, 1989.

No significant hazards consideration comments received: No

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

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: May 19, 1988

Brief description of amendment: The amendment clarifies and corrects minor problems and errors occurring in the Radiological Environmental Technical Specifications and clarifies the reporting requirements for major modifications to the radioactive waste systems.

Date of issuance: May 9, 1989

Effective date: May 9, 1989

Amendment No.: 127

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

Date of initial notice in **Federal Register**: March 8, 1989 (54 FR 9925). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 9, 1989.

No significant hazards consideration comments received: No

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

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 7, 1987, supplemented April 11, 1989

Brief description of amendment: The amendment clarifies the license conditions governing receipt, possession and use of radioactive materials such as apparatus, components and tools.

Date of issuance: May 9, 1989

Effective date: May 9, 1989

Amendment No.: 128

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

Date of initial notice in **Federal Register**: June 1, 1988 (53 FR 20045). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 9, 1989.

No significant hazards consideration comments received: No

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

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: February 2, 1989

Brief description of amendment: This amendment increased, on a one time basis, the 18 month surveillance interval by approximately 2 months for the A and D emergency diesel generators.

Date of issuance: May 15, 1989

Effective date: May 15, 1989

Amendment No.: 25

Facility Operating License No. NPF-57. This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 22, 1989 (54 FR 11841). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 1989.

No significant hazards consideration comments received: No

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

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

Date of application for amendments: December 20, 1988 and supplemented on March 3, 1989, to provide clarifications.

Brief description of amendments: Deleted the Residual Heat Removal System autoclosure interlock.

Date of issuance: May 2, 1989

Effective date: Unit 1 effective as of startup from the eighth refueling outage scheduled to end in May 1989; Unit 2 effective as of startup from the fifth refueling outage currently scheduled for March 1990.

Amendment Nos.: 95 and 71

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

Date of initial notice in **Federal Register**: February 8, 1989 (54 FR 6206). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 2, 1989.

No significant hazards consideration comments received: No

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

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

Date of application for amendments: December 30, 1988 and supplemented on April 19 and May 4, 1989. The supplemental letters provided corrected technical specification pages which did not change the technical requirements and a commitment to revise the FSAR.

Brief description of amendments: The amendments changed the technical specifications to permit the use of VANTAGE 5 Hybrid fuel, reduce flow measurement uncertainty allowances and eliminate the rod bow penalty factor.

Date of issuance: For Unit 1, the amendment is effective as of the date of issuance. For Unit 2, the amendment is effective as of fuel load during the fifth refueling outage currently scheduled to begin March 1990.

Effective date: May 9, 1989

Amendment Nos.: 96 and 72

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

Date of initial notice in **Federal Register:** February 8, 1989 (54 FR 6207). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 9, 1989.

No significant hazards consideration comments received: No

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

Southern California Edison Company, et al., Docket Nos. 50-206, 50-361 and 50-362, San Onofre Nuclear Generating Station, Units 1, 2 and 3, San Diego County, California

Date of application for amendments: February 14, 1989

Brief description of amendments: The amendments revise Technical Specification 6.2.2.g, "Unit Staff," to explicitly require the Assistant Plant Superintendent to maintain a senior reactor operator license.

Date of issuance: May 9, 1989

Effective date: May 9, 1989

Amendment Nos.: 126, 71 and 59

Facility Operating License Nos. DPR-13, NPF-10 and NPF-15: Amendments changed the Technical Specifications.

Date of initial notice in **Federal Register:** March 22, 1989 (54 FR 11842). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 9, 1989.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 11, 1989

Brief description of amendment: The amendment revised Technical Specification 3.5.1, "Reactor Trip Instrumentation," by including a footnote relating to Mode 2 of the Applicable Modes column for Function Unit 4, the Intermediate Range, Neutron Flux, in Table 3.5.1-1. The footnote indicates that the startup rate circuit for the intermediate range neutron flux channels will be enabled at 10^{-4} percent of full reactor power instead of at 10^{-6} percent of full reactor power as implied but not previously specified in the technical specifications.

Date of issuance: May 16, 1989

Effective date: This license amendment is effective the date of issuance.

Amendment No.: 128

Provisional Operating License No. DPR-13. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 28, 1989 (54 FR 18369). The notice stated that by May 30, 1989, the licensees may file a request for a hearing with respect to issuance of the amendment and any person whose interest may be affected by the proceeding, and who wishes to participate as a party must file a written request for a hearing and a petition for leave to intervene. The notice further stated that the amendment would not normally be issued until the expiration of the date above, but if circumstances should 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 notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The Commission's related evaluation of the amendment and its final determination of no significant hazards consideration are contained in a Safety Evaluation dated: May 16, 1989

No significant hazards consideration comments received: No comments.

Local Public Document Room location: General Library, University of California, Post Office Box 19557, Irvine, California 92713.

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

Date of application for amendment: December 22, 1988 (TS 264)

Brief description of amendment: The amendment adds surveillance requirements and time delays to the Reactor Protection System power monitoring system.

Date of issuance: May 16, 1989

Effective date: May 16, 1989, and shall be implemented within 60 days

Amendment No.: 164

Facility Operating License No. DPR-52: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 1, 1989 (54 FR 5176). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 16, 1989.

No significant hazards consideration comments received: No

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

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

Date of application for amendments: June 24, 1987 (TS 87-17)

Brief description of amendments: The amendments revise the Sequoyah Nuclear Plant, Units 1 and 2, Technical Specifications (TS). The changes are throughout the TS to correct thirty inconsistencies, minor discrepancies, factual errors and typographical errors within the TS. One change is to remove an error from a previous TS amendment. Twelve changes correct typographical errors. Four changes correct references to figures or the figure itself. Eight changes correct inconsistancies between the Unit 1 TS and the Unit 2 TS. Five changes correct factual errors. Correction of these errors will eliminate confusion over applicable requirements and the potential for error in reading the TS.

The proposed change to correct the alphabetical listing of the definitions in the index was approved in Amendment 71 for Unit 1 and Amendment 63 for Unit 2. These amendments were issued by letter dated May 18, 1988.

Date of issuance: May 5, 1989

Effective date: May 5, 1989

Amendment Nos.: 114, 104

Facility Operating Licenses Nos. DPR-77 and DPR-79. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 20, 1988 (53 FR 13021). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 5, 1989.

No significant hazards consideration comments received: No

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

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

Date of application for amendments: February 23, 1989 (TS 88-24/88-02)

Brief description of amendments: The amendments modify the Sequoyah Nuclear Plant, Units 1 and 2, Technical Sepcifications (TS). The changes revise the surveillance requirement 4.7.1.2.a to add specific, differential pressure test values for each auxiliary feedwater (AFW) pump. The associated bases section is revised to clarify the AFW technical specification requirements.

The changes for the Unit 2 TS superseded the values submitted in the licensee's application dated May 26, 1988 for TS change number 88-02. The new higher proposed differential pressure values for Unit 2 are to provide additional margin to offset uncertainties in the flow and pressure test data for the three AFW pumps. A revised bases for Unit 2 was also submitted.

Date of issuance: May 11, 1989

Effective date: May 11, 1989

Amendment Nos.: 115, 105

Facility Operating Licenses Nos. DPR-77 and DPR-79. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 22, 1989 (54 FR 11844). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 11, 1989.

No significant hazards consideration comments received: No

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

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

Date of application for amendment: January 14, 1986 as supplemented by letter dated April 14, 1989.

Brief description of amendment: The amendment incorporated Technical Specification limiting conditions for operation and surveillance requirements for the steam generator Atmospheric Steam Dumps (ASD's) into the Callaway Operating License in order to assure the availability of mitigating equipment assumed in the steam generator tube rupture analysis. The Technical Specification requirements constitute additional limitations on facility operations and satisfy, in part, the specific requirements of License Condition 2.C.(11) of the operating license.

Date of issuance: May 16, 1989

Effective date: May 16, 1989

Amendment No.: 45

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

Date of initial notice in **Federal Register:** August 13, 1986 (51 FR 29014). The April 14 and May 5, 1989 submittals provided additional clarifying information and did not change the proposed finding of the initial notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 16, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251 and the John M. Olin Library, Washington University, Skinker and Lindell Boulevards, St. Louis, Missouri 63130.

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

Date of application for amendments: October 19, 1984

Brief description of amendments: The amendments add surveillance requirements for the butterfly-type containment isolation valves in the containment purge lines and the containment vacuum ejector lines.

Date of issuance: May 8, 1989

Effective date: May 8, 1989

Amendment Nos.: 116 and 99

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

Date of initial notice in **Federal Register:** April 5, 1989 (54 FR 13770). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 8, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: The Alderman Library, Manuscripts Department, University of Virginia, Charlottesville, Virginia 22901.

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

Date of application for amendments: April 26, 1988, as supplemented July 18, 1988

Brief description of amendments: These amendments revise the Technical Specifications to allow entry into the containment personnel airlock during power operations to make repairs on the inner door of the personnel airlock. In addition, the definition of containment integrity has been revised to clarify the actions to be taken for inoperable automatic containment isolation valve(s).

Date of issuance: May 18, 1989

Effective date: May 18, 1989

Amendment Nos.: 126 and 126

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

Date of initial notice in **Federal Register:** August 24, 1988 (53 FR 32300). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 18, 1989.

No significant hazards consideration comments received: No

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

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

Date of application for amendments: January 6, 1987 as clarified April 14 and May 15, 1987.

Brief description of amendments: The amendments incorporated a change to Technical Specification Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels," clarifying the requirements for reactor coolant flow logic testing.

Date of issuance: May 18, 1989

Effective date: May 18, 1989

Amendment Nos.: 121 & 124

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

Date of initial notice in **Federal Register:** April 22, 1987 (52 FR 13353) and November 16, 1988 (53 FR 46165). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 18, 1989.

No significant hazards consideration comments received: No.

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

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

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

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

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

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

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

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

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Reactor Projects.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendments. By June 30, 1989, 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, DC 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. Where petitions are filed during the last 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 1-(800) 325-6000 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to (*Project Director*): petitioner's name and telephone number; date petition was mailed; plant name; and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

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

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

Date of application for amendment: April 28, 1989

Brief description of amendment: This amendment revises the Technical Specifications Section 3/4.6.4.1 to change the required Action statement when one of two redundant vacuum breaker position indicators is inoperable.

Date of Issuance: May 10, 1989

Effective date: May 10, 1989

Amendment No.: 32

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

Press release issued requesting comments as to proposed no significant hazards consideration: Yes, May 4, 1989 Monroe Evening News

Comments received: No.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated May 10, 1989.

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

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

NRC Project Director: Theodore R. Quay, Acting.

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

Date of Application for amendment: April 21, 1989

Brief description of amendment: The amendment changed Technical Specification Table 4.3-1 of the NRC record copy to agree with Table 4.3-1 of the distribution copy.

Date of Issuance: May 9, 1989

Effective Date: May 9, 1989

Amendment No.: 73

Facility Operating License No. DPR-75: Amendment revised the Technical Specifications.

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

The Commission's related evaluation of the amendment, consultation with the State of New Jersey and final no significant hazards considerations determination are contained in a Safety Evaluation dated May 9, 1989.

Attorney for licensee: Conner and Wetterhahn, 1747 Pennsylvania Avenue, Washington, DC 20006

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

NRC Project Director: Walter R. Butler

Dated at Rockville, Maryland, this 24th day of May, 1989.

For the Nuclear Regulatory Commission

Gary M. Holahan,

Acting Director, Division of Reactor Projects - III, IV, V and Special Projects Office of Nuclear Reactor Regulation

[Doc. 89-12889 Filed 5-3-89; 8:45 am]

BILLING CODE 7590-01-D