Stephen B. Bram
Vice President

Consolidated Edison Company of New York, Inc.
Indian Point Station
Broadway & Bleakley Avenue
Buchanan, NY 10511
Telephone (914) 737-8116

February 21, 1989

Re: Indian Point Unit No. 2
Docket No. 50-247

Document Control Desk
U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555

SUBJECT: Steam Generator Tube Inservice Inspection Program
(TAC No. 71778)

Attachment I to this letter serves to clarify our December 28, 1988 submittal entitled "Steam Generator Tube Inservice Inspection Program." Item 3 of that submittal entitled "Secondary Side Examination," has been expanded as per the verbal request made by your office on January 17, 1989.

If you have any further questions regarding this matter, please contact Mr. Jude G. Del Percio, Manager, Regulatory Affairs.

Very truly yours,

S B Bram

cc: Mr. William Russell
Regional Administrator - Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1498

Ms. Marylee M. Slosson, Project Manager
Project Directorate I-1
Division of Reactor Projects I/II
U.S. Nuclear Regulatory Commission
Mail Stop 14B-2
Washington, DC 20555

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 38
Buchanan, NY 10511

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Attachment I

Clarification to Item 3 of the Proposed
Steam Generator Tube Examination Program
1989 Refueling Outage

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
February, 1989

Tubes that do not pass the 610 mil probe will be plugged. Furthermore, the tubes immediately adjacent to any tube that does not pass the 610 mil probe also will be subjected to an eddy current examination.

The examination will be conducted from either the hot or cold leg side of the channel head. All tubes that require full length inspection will be examined over their full length from the mouth of the tube through the tubesheet, around the U-bend, to the mouth of the tube on the opposite side.

2. Flow Slot and Lower Support Plate Inspections

Using the hand holes above the tubesheet on all four steam generators, a visual and photographic examination of the lower tube support plates will be made. Where feasible, higher support plates also will be photographed through flow slots in the lower support plates.

Using the "hillside" inspection ports in Steam Generators 22 and 23, a visual and photographic examination will be made of the flow slots in the uppermost support plate.

3. Secondary Side Examination

A remote visual examination of the steam generator just above the tubesheet will be performed.

The examination will be conducted via CCTV, closed-circuit-television-camera, and documented on video tape. The tape will show the complete annulus at and immediately above the tubesheet level between the shell and the tube bundle; and may, based upon a best attempt, show the lanes between columns of tubes at and immediately above the tubesheet level through the tube bundle to the flow lane. Any debris that is located will be identified. Unusual conditions that are observed will be documented.

Foreign objects that are observed will be removed, if practical. If foreign objects are found and left in place, a justification for continued operation will be developed prior to return to service.

Some foreign objects that have been removed from the steam generators in the past were: pieces of wire, a 2 inch long pin, a 1-5/8 inch cotter pin and a small screw.

4. Steam Generator Sludge Analysis

The sludge that will be removed from the steam generator tubesheets during lancing operations will be sampled and chemically analyzed.

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DOCKET NO(S). 50-247

Mr. Stephen B. Bram
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadwayaand Bleakley Avenue, Buchanan, New York 10511

SUBJECT: Consolidated Edison Company of New York, Inc.

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 Feb. 8, 1989 [~~XXXXXXXXXX~~] March 10, 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

m/A-4

OFFICE	PDI-1 W						
SURNAME	CVogan						
DATE	2/21/89						

NRC FORM 318 (10/80) NRCM 0240 OFFICIAL RECORD COPY

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 January 16, 1989 through January 27, 1989. The last biweekly notice was published on February 1, 1989 (54 FR 5159).

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 March 10, 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.

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

Date of amendment request: June 30, 1988

Description of amendment request: The proposed amendment would modify the Technical Specifications by adding surveillance requirements for the automatic actuation of the shunt trip attachments of the reactor trip breakers, and for the silicon controlled rectifier (SCR) trip relays used to interrupt [illegible] to the control rods, as required by [illegible] 4.3 and 4.4 of Generic Letter 83-28, "Required Actions Based On Generic Implications of Salem Anticipated Transient Without Scram Events," and Generic Letter (GL) 85-10, "Technical Specifications For GL 83-28, Items 4.3 and 4.4."

This Notice supersedes the Notice published September 21, 1988 (53 FR 36668).

Basis for proposed no significant hazards consideration determination: 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 Commission has provided guidance for the application of the above criteria for no significant hazards consideration determination by providing examples of amendments that are considered not likely to involve significant hazards considerations (51 FR 7751). These examples include: Example (ii) A change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications: e.g., "a more stringent surveillance requirement."

The proposed addition of surveillance test requirements and limiting conditions for operation for the RTB shunt trip attachments and the SCR trip relays are additional limitations not presently included in the Technical Specifications, and are therefore within the scope of the example.

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

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

Attorney for licensee: Nicholas S. Reynolds, Esq., Bishop, Cook, Purcell, & Reynolds, 1400 L Street, NW., Washington, DC 20005-3502

NRC Project Director: Jose A. Calvo

Arkansas Power & Light Company, Docket No. 50-313 and 50-368, Arkansas Nuclear One, Units 1 and 2, Pope County, Arkansas

Date of amendment request: October 30, 1987

Description of amendment request: The proposed amendment would change the expiration date for Unit 1 Facility Operating License No. DPR-51 from December 6, 2008 to May 20, 2014 and would change the expiration date for Unit 2 Facility Operating License No. NPF-6, from December 6, 2012 to July 17, 2018.

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 Arkansas Power and Light Company reviewed the proposed change and determined for Unit 1 and Unit 2 that:

(1) The proposed change does not involve any changes in plant design, physical changes to plant systems, equipment or structures, or modifications to Technical Specifications or plant procedures. The original plant design provides for 40 years of operation and postulated accidents have been evaluated accordingly. Surveillance, inspection, testing, and maintenance programs are in place to sustain the condition of the plant throughout its service life. In conclusion, the potential effects of 40 years of operation have been considered in the existing design, analyses and operation of the plant and, therefore, the probability or consequences of previously evaluated accidents has not been significantly increased.

(2) Since the proposed change does not affect the design or operation of the plant and programs are in place to maintain the plant throughout its service life, the change does not increase the possibility of a new or different accident from those previously evaluated.

(3) The proposed change does not involve any changes in plant design, physical changes to plant systems, equipment or structures, or modifications to Technical Specifications or plant procedures. Existing surveillance, inspection, testing, and maintenance programs sustain the condition of the plant throughout its service life. These measures, together with continued operation in accordance with the Technical Specifications assure that an adequate margin of safety is preserved on a continuous basis. Therefore, the extension of the operating license term does not result in a significant reduction in a margin of safety.

Based on the previous discussion, the licensee concluded that the proposed amendment request does not involve a significant increase in the probability of a new or different kind of accident from any accident previously evaluated; nor involve a significant reduction in the required margin of safety. The NRC staff has reviewed the licensee's no significant hazards considerations determination and agrees with the licensee's analysis. The staff has, therefore, made a proposed determination that the licensee's request does not involve a significant hazards consideration.

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

Attorney for licensee: Nicholas S. Reynolds, Esq., Bishop, Cook, Purcell, & Reynolds, 1400 L Street, NW., Washington, DC 20005-3502

NRC Project Director: Jose A. Calvo

Connecticut Yankee Atomic Power Company, Docket No. 50-213, Haddam Neck Plant, Middlesex County, Connecticut and Northeast Nuclear Energy Company, et al., Docket Nos. 50-245/336/423, Millstone Nuclear Power Station, Unit Nos. 1, 2 and 3, New London County, Connecticut

Date of amendment request: January 12, 1989

Description of amendment request: The proposed amendments to the Technical Specifications (TS) will add a new requirement to TS Section 6.7, "Safety Limit Violation." This requirement will state that "operation shall not be resumed until authorized by the Commission." This proposed change will make the TS for the four plants consistent with the requirements of 10 CFR 50.36.

In addition a change to the Millstone Unit 3 TS has been proposed to change the requirement for auditing TS compliance from all provisions in each section to provision in each section, each year, during the five-year audit cycle for this plant. This change will make the Millstone Unit 3 TS consistent with the TS for the other three plants.

Basis for proposed no significant hazards consideration determination: The licensees have reviewed the proposed changes in accordance with 10 CFR 50.92 and have concluded and the staff agrees that they do not involve a significant hazards consideration in that these changes would not:

1. Involve a significant increase in the probability of occurrence or consequences of an accident previously analyzed. The proposed changes will make the technical specifications consistent with 10 CFR 50.36 for Safety Limit Violations. In addition, the change proposed in the area of Nuclear Review Board Audits will make Millstone Unit 3 consistent with the Westinghouse Standard TS and other Nuclear Plant's Technical Specifications. These changes will not increase the probability of occurrence or the consequences of an accident previously analyzed.

2. Create the possibility of a new or different kind of accident from any previously analyzed. Since there are no changes in the way the plant is operated, the potential for an unanalyzed accident is not created. No new failure modes are introduced.

3. Involve a significant reduction in the margin of safety. Since the proposed changes do not affect the consequences of any accident previously analyzed, there is no reduction in the 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: Russell Library, 123 Broad Street, Middletown, Connecticut 06457 and the Waterford Public Library, 49 Rope Ferry Road, Waterford, Connecticut 06385.

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 December 30, 1988 and revised January 20, 1989.

Description of amendment request: The amendment would revise the Indian Point 2 Technical Specifications to allow a fuel design transition to Westinghouse 15X15 Optimized Fuel Assemblies (OFA) fuel. Indian Point 2 has been operating with a Westinghouse 15X15 low-parasitic (LOPAR) fueled core. The

15X15 OFA fuel has design features similar to the 15X15 LOPAR fuel. The major design difference is the use of seven middle zircoloy grids for the OFA fuel versus seven middle inconel grids for the LOPAR fuel. Several of the plant operating limitations contained in the Technical Specifications will require revisions to allow the use of the OFA fuel and are discussed below.

1. *Administrative Changes* - References to LOPAR fuel throughout the Technical Specifications will be revised. In addition, the licensee is using this amendment application to delete obsolete requirements, relocate requirements to other sections of the Technical Specifications and make typographical corrections and clarifications.

2. *Improved Thermal Design Procedure and WRB-1 Correlation* - The proposed changes to Technical Specification Figure 2.1-1 would include a change to the thermal hydraulic design method used to satisfy the Departure from Nucleate Boiling design bases for Indian Point 2.

3. *Low Pressurizer Pressure Reactor Trip Setpoint* - The proposed changes to Technical Specification 2.3.1.B(3) would increase the minimum allowable value for the low pressurizer pressure reactor trip set point from greater than or equal to 1700 psig to greater than 1870 psig. The change is being proposed to revise the allowable setpoint trip to a value more consistent with plant operation.

4. *Over Temperature delta T and Over Power delta T Setpoints* - The proposed amendment would change Technical Specifications 2.3.1.B(4), 2.3.1.B(5) and Figure 2.1-1 concerning Overtemperature delta T and Overpower delta T. The revision to Figure 2.1-1 results from the implementation of a change in the Westinghouse DNB methodology as discussed in 2 above and a change in the allowable F delta H discussed in 8 below. The changes to 2.3.1.B(4) and 2.3.1.B(5) reflect the revised reactor core safety limits given in the proposed Figure 2.1-1.

5. *Boric Acid Storage System Volume* - The proposed revision to Section 3.2 changes the minimum volume requirements of the Boric Acid Storage System from 4400 to 6000 gallons. The revision is proposed to provide additional fuel management flexibility.

6. *Safety Injection Accumulators* - The proposed revision to Specification 3.3.A.1.C changes the Safety Injection Accumulators pressure and volume requirements from 600 psig and a minimum of 814.5 ft^3 and a maximum of 829.5 ft^3 to 615 psig and a minimum of 787.5 ft^3 and a maximum of 802.5 ft^3 respectively. The changes are proposed to provide increased flexibility in fuel management.

7. *Boron Concentration and Shutdown Margin* - The proposed revision to Specification 3.8.B.2 would decrease the required shutdown margin during refueling from 10% delta k/k to 5% delta k/k and fix the minimum refueling boron concentration at 2000 ppm. To maintain consistency Specification 3.6.A.1 will also be revised to reflect the revised shutdown margin and minimum boron concentration. The changes are proposed to provide increased flexibility in fuel management.

8. *Power Distribution F delta H* - The proposed revision to Specification 3.10.2.1 would increase the allowable peak value of F delta H at 100% power from 1.55 to 1.62. This change is proposed to increase flexibility in fuel management.

9. *Rod Drop Time* - The proposed revision to Specification 3.10.8, Rod Drop Time, would change the control rod drop time interval of 1.8 seconds from loss of stationary gripper core voltage to dashpot entry to a control rod drop time interval of 2.4 seconds from gripper release to dashpot entry.

10. *Hot Channel Factor $F_Q(Z)$* - The proposed revision to Technical Specification 3.10 would revise the normalized total peaking factor as a function of core height. This would increase the allowable normalized total peaking factor at the upper elevations of the reactor core and is being revised to reflect the new LOCA analyses.

11. *Low Pressure Safety Injection Setpoint* - The proposed revision to Table 3.5-1 changes the pressurizer low pressure safety injection setpoint from 1700 to 1829 psig. The purpose of the proposed change is to account for possible instrument error.

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 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.

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

1. Administrative changes

...these changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revisions do not affect plant operations. The proposed revisions delete obsolete specifications, relocate existing specifications and add corrections and clarifications.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed revisions delete obsolete specifications, relocate existing specifications and add corrections and clarifications. The proposed changes do not modify the plant's configuration or operation. Nothing would be added or removed that could conceivably introduce a new or different kind of accident mechanism or initiating circumstances than those previously evaluated.

(3) Involve a significant reduction in a margin 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.

2. Improve Thermal Design Procedure and WRB-1 Correlation

...this change would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The ITDP and WRB-1 represent changes to analyses methods only. The probability of an accident occurring is not impacted by the methods selected to evaluate the DNB design basis associated with that accident once it has been postulated to occur. The consequences of the accident must satisfy the same DNB design basis as previously evaluated. Use of ITDP and the WRB-1 do not decrease the available DNB margins when evaluating an accident.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The noted changes are to the methods used in evaluating the DNB design basis only and are involved in analyses only after an accident has been postulated to occur.

(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 DNB design criteria continues to be satisfied with the use of ITDP and the WRB-1. As described in the safety assessment, use of this improved method and correlation do not decrease DNB margin over methods and correlations previously used in Indian Point Unit 2.

3. Low Pressurizer Pressure Reactor Trip Setpoint

...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 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 allowable setpoint for reactor trip on low pressurizer pressure 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 could conceivably introduce a new or different kind of accident mechanism or initiating circumstance than that previously evaluated.

In general, the proposed change does not adversely effect the ability of the pressurizer low pressure reactor trip signal to perform its safety function to initiate reactor core shutdown during a rapid depressurization event.

(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 reactor trip on low pressurizer pressure is to initiate reactor core shutdown during a severe depressurization event and to ensure that the reactor coolant system pressure does not exceed the applicable lower limit for the overtemperature and overpower delta T protection. Worst case large and small break LOCA transients were reanalyzed using the latest approved computer codes and methodology as a basis for evaluating this proposed change. For the Non-LOCA accidents, analyses and evaluations demonstrate continued conformance to all applicable design and safety criteria.

4. Over Temperature delta T and Overpower delta T Setpoints

...these changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revision is supported by conservative evaluations 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 changes to the OT delta T and OP 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 could conceivably introduce a new or different kind of accident mechanism or initiating circumstance than that previously evaluated.

In general, the proposed changes do not adversely affect the ability of OT delta T and OP 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 still 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. Evaluations and/or analyses for all of the licensing basis accidents described in FSAR Chapter 14 which take credit for an OT delta T or OP delta T reactor trip have been performed and the results of these analyses and evaluations have demonstrated conformance with the applicable design and regulatory requirements.

5. Boric Acid Storage System Volume

...this change would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The volume or boric acid required in the boric acid storage system is not considered in the mitigation of Chapter 14 events. The volume is required to ensure that a sufficient volume of boric acid solution is available to borate the reactor coolant system to a cold shutdown condition.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The larger volume requirement is well within the capacity of the boric acid storage system. The RWST provides an alternative source of boric acid to meet redundancy requirements.

(3) Involve a significant reduction in a margin of safety. The use of the more conservative shutdown margin assumptions have not decreased, but actually increased cold shutdown boration capability.

6. Safety Injection Accumulators

...this change would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revisions are supported by conservative analysis utilizing the latest approved computer codes and methodology for large break LOCA and by evaluation of conformance to the applicable design and regulatory criteria in the unlikely event of a small or large break LOCA.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes to the accumulator cover gas pressure and water volume 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 could conceivably introduce a new or different kind of accident mechanism or initiating circumstances than those previously evaluated.

The proposed changes are within the capabilities of the system and do not adversely effect the ability of the emergency core cooling system accumulators to perform their safety function to provide passive injection of borated water to the reactor coolant system.

(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 emergency core cooling system accumulators is to provide passive injection of borated water to the reactor coolant system in the event of massive depressurization and loss of reactor coolant inventory. The worst case large break LOCA transient was reanalyzed using the latest approved computer codes and methodology as a basis for evaluating these proposed changes, and evaluations have determined that these changes will not adversely affect the results of small break LOCA analyses. These analyses/evaluations demonstrate continued conformance to all applicable design and safety criteria.

7. Boron Concentration Shutdown Margin

...these changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revision is supported by conservative analyses utilizing approved 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 refueling shutdown margin and minimum boron concentration 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 could conceivably introduce a new or different kind of accident mechanism or initiating circumstance than that previously evaluated.

In general, the proposed change does not adversely affect the ability to keep the reactor safely shutdown during refueling operations.

(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 refueling shutdown margin and minimum boron concentration is to keep the reactor core shutdown during refueling operations. Safety analyses for the licensing basis accident described in FSAR Chapter 14 which take credit for refueling boron concentration have been performed and the results of these analyses have demonstrated conformace with the applicable design and regulatory requirements.

8. Power Distribution F delta H

...this change would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The peak F delta H value represents a design limit on peaking factors which must be satisfied for plant operation. 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 and operational limit value of F delta H does not modify the plant's configuration or operation, and therefore the previously postulated accidents are the only ones that require evaluation or resolution. Nothing would be added or removed that could 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 changes, all safety criteria previously evaluated are still 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 LOCA and non-LOCA design and safety criteria continue to be satisfied including the impact of an increased F delta H.

9. Rod Drop Time

...these changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed revision is supported by conservative evaluations 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 control rod drop time for reactor trip 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 could conceivably introduce a new or different kind of accident mechanism or initiating circumstance than that previously evaluated.

In general, the proposed change does not adversely affect the ability of control rods to perform their safety function of intitiating core shutdown in response to a reactor trip signal.

(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 control rod drop in response to a reactor trip signal is to initiate reactor core shutdown. Safety evaluations and analysis for all of the licensing basis accidents described in FSAR Chapter 14 which take credit for a reactor trip have been performed and the results of these analyses and evaluations have demonstrated conformance with the applicable design and regulatory requirements.

10. Hot Channel Factor $F_Q(Z)$

...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 analyses utilizing the latest approved computer codes and methodology. These analyses have demonstrated conformance to the applicable design and regulatory criteria in the unlikely event of a small or large break LOCA.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change to the allowable core axial power distribution limits 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 could 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 previously evaluated are still met, remain conservative, and continue to maintain the previous margins of safety.

Worst case large and small break LOCA transients were reanalyzed using the latest approved computer codes and methodology as a basis for evaluating this proposed change. These anlayses demonstrate continued conformance to all applicable design and safety criteria.

11. Low Pressurizer Pressure Safety Injection Setpoint

...these changes would not:

(1) Involve a significant increase in the probability or consequence or an accident previously evaluated. The proposed revision assures that assumptions are met for the existing safety analyses.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change to the minimum allowable setpoint for safety injection on low pressurizer pressure 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 could conceivably introduce a new or different kind of accident mechanism or initiating circumstances than those previously evaluated.

(3) Involve a significant reduciton 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 safety injection on low pressurizer pressure is to initiate safety injection flow during a severe depressurization event. The proposed change will increase the allowable pressure setpoint and assure that safety injection flow will be delivered to the reactor core as assumed in the safety analyses.

The proposed changes to the Technical Specifications are as a result of core reload and not because of any significant changes made to the acceptance criteria for technical specifications, and the analytical methods used by the licensee in the required reload analyses have been previously found acceptable by the NRC. Therefore, based on the above the staff proposes that the proposed changes do not represent 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, Director

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 amendment request: December 21, 1987

Description of amendment request: The proposed amendments to Technical Specification (TS) 4.7.6.d would extend the sampling interval of the carbon adsorbers of the Control Room Area Ventilation System from 720 hours to 1440 hours. With sampling every 720 hours, the six sampling canisters in each of the two carbon beds would be used up in a year. Installing fresh sample canisters requires opening and resealing the covers of the carbon beds. By TS 4.7.6e, these operations require leak tests and penetration tests which normally would not be required for 18 months. By extending the sampling interval from 720 hours to 1440 hours, the surveillance required by TS 4.7.6e would need to be performed only after the normal 18-month interval. The state of the art triethylenediamine-treated carbon adsorbers have been demonstrated by laboratory tests to remain highly efficient in adsorbing methyl iodide after extended operation.

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; 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 December 21, 1987, provided the following discussion and anlysis with regard to the three 10 CFR 50.92 standards:

The OPERABILITY of the Control Room Area Ventilation System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. Technical Specification 4.7.6, Control Room Area Ventilation System Surveillance Requirements, ensures that the System remains operable as required.

Proposed Technical Specification 4.7.6.d seeks to extend the Control Room Ventilation System Carbon adsorber sample time interval from 720 hours to 1440 hours because existing requirements are overly restrictive. Catawba Nuclear Station is equipped with state of the art Control Room Ventilation System Carbon adsorbers which retain very high efficiencies over prolonged intervals of operation. Laboratory data support the efficiency of the Carbon adsorbers. Therefore, it is reasonable and justifiable to extend the carbon adsorber sample time interval as indicated in the proposed Technical Specification.

Existing Technical Specification 4.7.6.d indicates that each Control Room Area Ventilation System is to be demonstrated operable after every 720 hours (30 days) of carbon adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample meets the laboratory testing criteria of Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%. The filter units in service at Catawba Nuclear Station currently have no bypass mode. Therefore, either A train (1CRA-PFT-1) or B train (2CRA-PFT-1) must operate in the filtered mode continuously. This design configuration allows one unit to run continuously for 30 days before a sample must be removed.

Each filter unit is initially provided with six installed sample canisters. If one canister is removed every 30 days (one canister from each unit is removed every 2 months) the samples would be depleted after one year. The removed canisters are to be reloaded and reinstalled in the filter unit. Removal of the cover from the carbon bed jeopardizes Unit integrity and a refrigerant penetration leak rate test is required on the carbon bed whenever the cover is removed. This results in the Technical Specification 4.7.6.e surveillance test interval being reduced from the normal 18 month to one year or less. Therefore the existing sampling interval is overly restrictive and results in excessive sampling of the Control Room pressurizing filter units. Proposed Technical Specification 4.7.6.d would allow for a normal 18 month surveillance test interval (as required by existing Technical Specification 4.7.6.e) by extending the Technical Specification 4.7.6.d sample interval from 720 hours to 1440 hours.

Historical data supports the proposed Technical Specification 4.7.6.d sampling interval of 1440 hours. Laboratory sample analysis results for filter units 1CRA-PFT-1 and 2CRA-PFT-2 show that over the course of one year and more than 4,000 hours of run time per unit covering typical atmospheric and seasonal meteorological conditions, there was no noticeable degradation in the methly iodide efficiency of the carbon. The sample results varied from 99.98% to 99.95% for 1CRA-PFT-1 and from 99.99% to 99.90% efficiency for 2CRA-PFT-1. Therefore, the proposed extension of the Technical Specifications 4.7.6.d sample interval is justifiable due to the high efficiency of the carbon in 1CRA-PFT-1 and 2CRA-PFT-1 and their ability to retain their efficiency over the course of prolonged operation as shown by the subject laboratory sample results.

The air flow rate through 1CRA-PFT-1 and 2CRA-PFT-2 is 6,000 cubic feet per minute (CFM) of which 4,000 cfm is outside air and 2,000 is recirculated Control Room area air. Since Catawba Nuclear Station is located in a rural environment, away from any major industrial plants, the outside air is essentially clean and free of any industrial pollutants. Therefore, circulation of outside air through the filter units has no detrimental effect on the efficiency of the carbon. This phenomena is demonstrated by Catawba's carbon analyses results from the start of plant operation.

Additionally, the carbon utilized at Catawba Nuclear Station is activated and impregnated with Triethylenediamine (TEDA). This type of carbon is a state-of-art-the-art material which results in high methyl iodide efficiency as shown by laboratory analysis of the samples. The 720 hours run time interval recommended by Regulatory Guide 1.52 is an arbitrary value applying to activated carbon. Since Catawba's carbon is activated impregnated with TEDA, the methyl iodide efficiency has been increased substantially.

In summary, the Control Room Ventilation System carbon adsorbers have been proven to maintain very high levels of Methyl Iodide efficiency under extended operation conditions. Laboratory analysis of carbon samples indicate that extending the sampling interval to 1440 hours has an insignificant effect on the efficiency of the adsorbers. Also, outside air circulated through the adsorbers is of high quality and would not impact the efficiency of the adsorbers even if sampling intervals are extended. Therefore, the proposed change to Technical Specification 4.7.6.d is reasonable and technically justifiable.

Pursuant to 10 CFR 50.92, this analysis provides a determination that the proposed amendments to the Technical Specifications involves no significant hazards considerations if operation 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 amendment does not involve a significant increase in the probability or consequences of any previously evaluated accident. Catawba Nuclear Station is equipped with state of the art Control Room Ventilation System Carbon adsorbers which retain very high Methyl Iodide efficiencies over prolonged intervals of operation. Previous laboratory analysis results indicate that over the course of a one year and more than 4,000 hours of runtime covering typical atmospheric and seasonal conditions, there is no noticeable degradation in the methyl iodide efficiency of the adsorber and that the carbon is perfectly capable of extended operation. Increasing the Technical Specification 4.7.6.d sample time intervals to 1440 hours has no significant impact to the efficiency of the carbon adsorbers and Control Room Area Ventilation System operability. Therefore, the proposed change cannot increase the probability or consequences of any previously evaluated accident.

The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed increase of Technical Specification 4.7.6.d sample time intervals to 1440 hours has no effect on the function, operation, or efficiency of the Control Room Area Ventilation System. Therefore the proposed Technical Specification change cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment does not involve a significant reduction in a margin of safety. As it was previously indicated, the Catawba Station Control Room Ventilation System Carbon adsorbers are capable of extended operation without any significant reduction in their Methyl Iodide removal efficiency. Previous laboratory carbon sample analysis results indicate that the proposed carbon adsorber sampling interval of 1440 hours will not reduce the efficiency of the Control Room Ventilation System in any significant manner. Therefore, the proposed Technical Specification does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's submittal and agrees that the proposed amendments would not have a significant adverse effect on the safe operation of the facility. Laboratory tests and plant experience have demonstrated the continued high adsorption efficiency of triethylenediamine-treated carbon after air ciculuation for 4000 hours. Also, contaminants such as industrial pollutants which could affect the carbon adsorption efficiency are absent in the pure outside air in the rural environment of the plant. The staff also agrees with the licensee's evaluation of the proposed amendment with respect to the three standards of 10 CFR 50.92.

On this basis, the Commission has concluded 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: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

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

NRC Project Director: David B. Matthews

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 amendment request: September 19, 1988, as supplemented December 28, 1988

Description of amendment request: The proposed amendments would revise the setpoints for Catawba Unit 2 steam generator level trips due to the planned relocation of level taps. The changes are applicable to Unit 2 only. Unit 1 is included administratively because the Technical Specifications are combined in one document for both units. The proposed changes for Unit 2 would revise:

(1) Table 2.2-1, Item 13.b.

(2) Table 3.3-4, Items 5.b.2., 6.c.2., and 8.c.2)

(3) the basis for Steam Generator Water Level, page B 2-7

Catawba Unit 2 is equipped with Westinghouse Model D5 steam generators while Unit 1 has Model D3. A major difference between those two models is the design of the moisture separator section. Two aspects of this design difference are of significance with respect to the proposed modification: (1) The D5 has a higher recirculation rate than the D3, and (2) the elevation of the lower deck plate in the D5 is higher than in the D3. Due to these differences, the lower instrument tap for the narrow range level instrumentation was located above the transition cone and lower deck plate on the D5 as opposed to below the transition cone in the downcomer in the D3. This has resulted in significantly different operating characteristics. The proposed modification will relocate the D5 lower instrument tap to the same location as the D3. Due to the location of the lower tap in model D5 generators, the shrink and swell characteristics are more pronounced than in the D3 model. This makes plant control difficult and more susceptible to trips.

In order to determine the potential gain in operational control characteristics of the D5 steam generator if the lower instrument tap were relocated to the equivalent location as the D3, Duke and Westinghouse installed pilot instrumentation on the Catawba 2C generator. Transient data have shown that the modified D5 level instrumentation will perform similarly to the D3 in terms of post-trip response.

The present span between the high level and low level trips on the D5 is physically bounded by the elevation of the top of the moisture separator swirl vanes and the elevation of the lower instrument tap, respectively. By relocating the lower tap, the lower level trip setpoint can be reduced. The high and operating level trip setpoints will also be reduced. The low level trip setpoint will be set at the elevation of the lower deck plate. With this arrangement, the margin between the operating level setpoint and low level trip setpoint will be increased from a current 42" to 58". This will make Unit 2 more tolerant to feedwater system malfunctions at power, thus reducing unnecessary trips and corresponding challenges to safety systems.

Relocating the narrow range instrumentation lower sensing tap on the Westinghouse model D5 steam generators to the same elevation as the model D3 steam generators would provide the following safety enhancements:

(1) The effects of level shrink and swell at low power levels will be greatly reduced, thus reducing the potential for reactor trips.

(2) The time necessary to recover indicated level following a reactor trip will be greatly reduced, thus reducing the potential for an overcooling event due to excessive auxiliary feedwater.

(3) The margin to low level trip will be increased thus reducing the potential for reactor trips at power.

Relocation of the level sensing tap to the downcomer region requires that the velocity induced error be accounted for in the determination of trip and operating level setpoints. This can be accomplished without reducing any current margin to trip.

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 proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated because the Westinghouse Safety Evaluation discusses the transients not requiring any reanalysis as well as those that required reanalysis. Its findings indicated that no conclusions in the Catawba Final Safety Analysis Report will be violated by relocating the steam generator level taps. The licensee reviewed two other events: (1) steam generator tube rupture and (2) loss-of-coolant accident (LOCA) to evaluate the need for reanalyses. The licensee concluded that no reanalyses were needed and that the conclusions of the current analyses remain bounding.

The proposed amendments do not create the possibility of a new or different kind of accident from any accident previously evaluated because relocating the level tap on the D5 generator should improve operation and no new modes of operation are introduced.

The proposed amendments do not involve a significant reduction in margin of safety because the modification would enhance safety by making the steam generators less susceptible to feedwater transients. This would reduce the potential for reactor and turbine trips and would avoid unnecessary transients on the primary and secondary systems.

Accordingly, the Commission has concluded that the requested changes meet the three standards and, therefore, has made a proposed determination that the requested license amendments do not involve a significant hazards consideration.

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

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

NRC Project Director: David B. Matthews

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

Date of amendment request: November 28, 1988

Description of amendment request: The proposed amendments would change Technical Specification (TS) TS 6.2.3 to clarify and supplement the specified function, composition, responsibilities, reporting, and records requirements for the Catawba Safety Review Group (CSRG) consistent with Item I.B.1.2 of NUREG-0737. Specifically,

- The function of the CSRG in TS 6.2.3.1 would be revised to specifically define the function of the group.
- The composition of the CSRG in TS 6.2.3.2 would be revised to add the qualification requirements for members of the group.
- The responsibilities requirement of TS 6.2.3.3 would be revised to replace a general statement with an itemized list of specific responsibilities.
- The reporting of the CSRG, specified by TS 6.2.3.4, would be revised to reflect that they report to the Manager of Nuclear Safety Assurance, rather than

to the Director, Nuclear Safety Review Board.

- The recordkeeping and distribution requirements of TS 6.2.3.5 would be revised to require that records of CSRG activities be maintained for the life of the station, and that reports of CSRG activities be forwarded to the Manager of Nuclear Safety Assurance.

Basis for proposed no significant hazards consideration determination: TS 6.2.3 provides requirements regarding administrative controls for the CSRG which represents the "Independent Safety Engineering Group" required by Item I.B.1.2 of NUREG-0737. The existing TS 6.2.3 is ambiguous and lacking in the necessary level of specificity to ensure effective control regarding the function, composition, responsibilities, reporting and records requirements of the CSRG. The proposed changes would correct this deficiency and, thereby provide increased assurance of compliance with Item I.B.1.2 of NUREG-0737.

The Commission has provided certain examples (51 FR 7744) of actions likely to involve no significant hazards considerations. The proposed changes do not match the examples. However, the staff has reviewed the licensee's request for amendments and has determined that should this request be implemented, it 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; rather these changes ensure that the administrative control aspects for the CSRG will be maintained in accordance with NUREG-0737 requirements for an independent safety engineering group. Accordingly, the Commission proposes to find that the changes do not involve a significant hazards consideration.

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

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

NRC Project Director: David B. Matthews

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 amendment request: January 17, 1989

Description of amendment request: The proposed amendments would revise Technical Specification (TS) Table 3.3-5, Item 15, and Surveillance Requirement (SR) 4.7.1.2.1b.4) to increase the Auxiliary Feedwater (CA) System suction swapover time from less than or equal to 15 seconds to less than or equal to 16 seconds. This would be accomplished by increasing the delay time from 5 to a maximum of 6 seconds. The proposed wording of the notes associated with Item 15 of TS Table 3.3-5 and SR 4.7.1.2.1b.4) would be modified to clearly state that the 6 seconds represent the maximum delay time and that a shorter delay may be acceptable.

This proposed change is in response to Corrective Action (9) contained in Licensee Event Report (LER) 414/88-12 dated April 8, 1988. This LER described an incident at Catawba Unit 2 where Train A Suction for the motor-driven CA pump inadvertently swapped over from the normal condensate grade supply to the Nuclear Service Water System.

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 evaluted; or (3) involve a significant reduction in a margin of safety.

The proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed changes would reduce the probability of an inadvertent swapover and would not affect the previously evaluated accident analyses discussed in the Final Safety Analysis Report.

The proposed amendments do not create the possibility of a new or different kind of accident from any accident previously evaluated because the increase in the swapover time would not significantly impact the design basis of the system and no new modes of operation are introduced.

The proposed amendments do not involve a significant reduction in a margin of safety because the changes would reduce the probability of an inadvertent swapover without increasing its consequences.

Accordingly, the Commission has concluded that the requested changes meet the three standards and, therefore, has made a proposed determination that the requested license amendments do not involve a significant hazards consideration.

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

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

NRC Project Director: David B. Matthews

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

Date of amendment request: November 28, 1988

Description of amendment request: The proposed amendments would change the name of the "Station Safety Review Group (SSRG)" in Technical Specification (TS) 6.2.3 to the "McGuire Safety Review Group (MSRG)." The change to TS 6.2.3 would also clarify and supplement the specified function, composition, responsibilities, reporting, and records requirements for the MSRG consistent with Item I.B.I.2 of NUREG-0737. Specifically,

- The function of the MSRG in TS 6.2.3.1 would be revised to specifically define the function of the group.

- The composition of the MSRG in TS 6.2.3.1 would be revised to add the qualification requirements for members of the group.

- The responsibilities requirement of TS 6.2.3.3 would be revised to replace a general statement with an itemized list of specific responsibilities.

- The reporting of the MSRG, specified by TS 6.2.3.4, would be revised to reflect that they report to the Manager of Nuclear Safety Assurance, rather than to the Director, Nuclear Safety Review Board.

- The recordkeeping and distribution requirements of TS 6.2.3.5 would be revised to require that records of MSRG activities be maintained for the life of the station, and that reports of MSRG activities be forwarded to the Manager of Nuclear Safety Assurance.

Basis for proposed no significant hazards consideration determination: TS 6.2.3 provides requirements regarding administrative controls for the MSRG. The MSRG at McGuire represents the "Independent Safety Engineering Group" which is required by Item I.B.I.2 of NUREG-0737. The existing TS 6.2.3 is ambiguous and lacking in the necessary level of specificity to ensure effective control regarding the function, composition, responsibilities, reporting and records requirements of the MSRG.

The proposed changes would correct this deficiency and, thereby provide increased assurance of compliance with NUREG-0737 Item I.B.1.2.

The Commission has provided certain examples (51 FR 7744) of actions likely to involve no significant hazards considerations. One of the examples (i) is "a purely administrative change to technical specifications; for example, ... a change in nomenclature." The change to replace SSRG by MSRG matches this example. The other proposed changes do not match the examples. However, the staff has reviewed the licensee's request for amendments and has determined that should this request be implemented, it 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; rather these changes ensure that the administrative control aspects for the MSRG will be maintained in accordance with the NUREG-0737 requirements for an independent safety engineering group. Accordingly, the Commission proposes to find that the changes do not involve a significant hazards consideration.

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

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

NRC Project Director: David B. Matthews

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

Date of amendment request: December 23, 1988

Description of amendment request: The amendment would allow the licensee to store fuel of up to 4.5 percent enrichment in both the dry fuel storage racks and storage pool A. This request is a result of the licensee's intent to use fuel of up to 4.2 percent enrichment during Fuel Cycle 9. The licensee is currently limited to storing fuel of 4.0 and 3.5 percent enrichment in the dry fuel storage racks and storage pool A respectively.

Basis for proposed no significant hazards consideration determination: The Commission has provided criteria 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 addressed the above three criteria in the amendment application and made a no significant hazards consideration determination. In regard to the first criterion the licensee provided the following analysis:

This amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

An increase in fuel enrichment will not by itself affect the mixture of fission product nuclides. A change in fuel cycle design which makes use of an increased enrichment may result in fuel burnup consisting of a somewhat different mixture of nuclides. The effect of this instance is insignificant because:

(a) The isotopic mixture of the irradiated assembly is relatively insensitive to the assembly's initial enrichment.

(b) Most accident doses are such a small fraction of 10 CFR Part 100 limits, a large margin exists before any change becomes significant.

(c) The change in Pu content which would result from an increase in burnup would produce more of some fission product nuclides and less of other nuclides. Small increases in some doses are offset by reductions in other doses. The radiological consequences of accidents are not significantly changed.

With respect to the second criterion the licensee stated:

This amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

As indicated in the enclosed analyses, an unplanned criticality event will not occur as Keff will not exceed 0.95 with the maximum allowable enriched fuel in pool A, and flooded with unborated water, or the dry storage racks immersed in a water mist of 7.5% moderator density. Criticality is possible for a mist environment only if the higher enriched fuel occupies all of the locations in the dry storage racks including those which are required to be vacant. To prevent [this] occurrence, FPC commits to establish controls to preclude improper fuel storage.

In regards to the third criterion the licensee provides the following statement:

This amendment will not involve a significant reduction in a margin of safety.

While the increased enrichment in pool A and the dry storage racks may lessen the margin to criticality this reduction is not significant because the overall safety margin is within NRC criteria of Keff [less than] 0.95 (NRC Standard Review Plan, Section 9.2.1).

Therefore, this amendment request satisfies the criteria specified in 10 CFR 50.92 for amendments which do not involve a significant hazards consideration.

The staff has reviewed the analysis provided by the licensee in support of a no significant hazards consideration determination. The staff agrees with the licensee's analysis and believes that the licensee has met the criteria for such a determination. Therefore, the staff proposes to determine that the proposed change does not involve a significant hazards consideration.

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

Attorney for licensee: R. W. Neiser, Senior Vice President and General Counsel, Florida Power Corporation, P. O. Box 14042, St. Petersburg, Florida 33733

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: December 4, 1987

Description of amendment request: The proposed amendment would revise TMI-2 Operating License No. DPR-73 by modifying Appendix A Technical Specifications Sections 1.13 - Definitions, and 3 - Limiting Conditions for Operation. The proposed amendment would revise the specifications related to fire protection systems at TMI-2. The proposed changes would align license requirements of fire protection systems consistent with the current, as well as future plant conditions through the remainder of the current cleanup operations.

A revised definition of "Fire Suppression Water System", Section 1.13, is proposed. The definition describes the components of the fire suppression water system. The revised definition deletes the terms "sprinkler" and "spray system riser" to be consistent with the revised requirements of Technical Specification 3.7.10.2 Deluge/Sprinkler Systems.

The licensee proposes to revise Technical Specification 3.7.10.1, Fire Suppression Water System, by eliminating one of four separate and redundant high pressure fire pumps and one of four separate water supplies supplying water to the pumps. The licensee further proposes to delete the requirement to maintain operability of the Unit 2 River Water Intake Diesel Fire Pump and the Unit 2 River Water

Intake Structure. The licensee also proposes to remove the terms "sprinkler" and "spray system user" to be consistent with the revised requirements of Technical Specification 3.7.10.2 - Deluge/Sprinkler Systems.

Technical Specification 3.7.10.2 - Deluge/Sprinkler Systems, would be deleted by the licensee. The current Technical Specifications require deluge and/or sprinkler systems in a number of areas in the TMI-2 ventilation system. The purpose of this system is for suppression of charcoal filter fires in the ventilation system. The licensee has determined that the ventilation system is no longer necessary to maintain the safe shutdown condition of the plant or to maintain off-site doses to less than 10 CFR Part 100 limits.

Section 3.7.10.3 - Halon System, requires that the Halon systems in the Cable and Transformer Rooms and four zones of the air intake tunnel be operable. The licensee proposes to delete this system. The licensee has determined that these areas, protected by the Halon system and located outside the Reactor Building, would not affect the safe shutdown condition of the plant nor would it result in an off-site release greater than 10 CFR Part 100 limits.

Technical Specifications 3.7.11 - Penetration Fire Barriers, would be deleted by the licensee. The current Technical Specification requires that all penetration fire barriers protecting safety related areas be functional. The November 17, 1987 revised Fire Protection Program Evaluation establishes the Reactor Building as the only fire area. The licensee has determined that maintenance of the penetration fire barriers are not necessary to ensure the safe shutdown of the facility.

The licensee proposes to modify the Bases Section 3/4.3.3.8 and 3/4.3.3.9 - Fire Detection Instrumentation, by making reference to the TMI-2 Fire Protection Program Evaluation with regard to adequate fire warning capability, delete reference to safety related equipment and permit remote surveillance techniques in lieu of fire patrols when fire detection instrumentation is inoperative.

Bases Section 3/4.7.10 - Fire Suppression Systems, would be similarly modified making reference to the TMI-2 Fire Protection Program Evaluation with regard to adequate fire suppression capability, and delete reference to the fire suppression system capability to minimize potential damage to safety related equipment. The Basis currently refers to four main fire pumps in the fire suppression system. Consistent with the changes proposed in Section 3.7.10.1, the Basis would be changed to refer to three main fire pumps. The licensee also proposes to delete the reference to the necessity for immediate corrective measures should the fire suppression water system become inoperative. The licensee proposes instead to state that the inoperability of the system would not affect the capability to maintain the safe shutdown condition of the plant nor the capability to prevent off-site releases greater than 10 CFR Part 100 limits. The licensee would retain the statement that if portions of the fire suppression system are inoperable, alternate backup fire fighting equipment would be made available in affected areas until the affected equipment could be restored to service.

The licensee proposes to delete Basis Section 3/4.7.11 Penetration Fire Barriers, consistent with the request to delete Section 3.7.11 - Penetration Fire Barriers.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists 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, (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.

TMI-2 is currently in a post-accident, cold shutdown, long-term cleanup mode, with sufficient decay heat removal assured by direct heat loss from the reactor coolant system to the reactor building atmosphere. The licensee is presently engaged in defueling the damaged reactor, decontaminating the facility and readying the plant for long-term storage. As of the end of December 1988, approximately 70 percent of the fuel contained in the reactor vessel has been removed. Defueling the facility has progressed to the regions below the location of the original core volume. Defueling activities within the reactor building will be completed by fall of 1989. The staff has determined in previous license amendments, that the potential accidents analyzed for TMI-2 in the current cleanup-mode are bounded in scope and severity by the range of accidents originally analyzed in the facility FSAR. The changes proposed by the licensee are changes to the Appendix A Technical Specifications reducing the fire protection requirements necessary to assure the safe shutdown of the facility. Since the facility is in a safe shutdown configuration, the reactor system is not pressurized and the core is partially defueled the licensee asserts that a reduction in fire protection measures is warranted and that off-site doses, even in the event of a fire, would be less than 10 CFR Part 100 limits.

The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated. The proposed changes to the Technical Specifications are based on a safety analysis contained in the November 17, 1987 Fire Protection Plan Evaluation (FPPE) submitted by the licensee in support of the proposed changes. The FPPE concludes that maintenance of only one fire area, the TMI-2 Reactor Building, is justified and that this assumption will not affect either the capability to maintain the monitored safe shutdown condition of the plant nor result in off-site doses greater than 10 CFR Part 100 limits.

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because no new modes of operation or new equipment are being introduced. The proposed changes do not involve a significant reduction in a margin of safety since the changes are consistent with the results of the recent Fire Protection Program Evaluation and do not affect the capability of the licensee to maintain the safe shutdown condition of the facility nor result in the possibility of off-site doses greater than 10 CFR Part 100 limits. The proposed changes will still require fire detection and suppression capability in the Reactor Building.

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

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

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts, & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: John F. Stolz

Louisiana Power and Light Company, Docket No. 50-382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: December 23, 1988

Description of amendment request: The proposed amendment would change the Technical Specifications to show a new location for one of the backup seismic monitors.

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, (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 seismic monitor being moved is currently in a location to detect peak accelerations on the reactor coolant system (RCS) piping by being attached to a pipe directly connected to the RCS piping. The location, however, is in a harsh environment subject to vibrations from a reactor coolant pump. The environment and vibrations continually damage the monitor rendering it useless. The new location will also place the monitor on a connected pipe to the RCS and should provide comparable information with less chance of unrelated damage. These backup seismic monitors do not influence any accident previously evaluated except possibly for the small added weight of the monitor on the connecting pipe. The licensee has evaluated the effects of the added weight at the new location; the weight does not cause the new seismic stress valves to exceed any limits. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

The monitor provides backup information to verify seismic induced stress calculations. It is not powered by external power sources and the weight at the new location should have no effect on the piping. The mounting clamp and monitor meet seismic Category 1 requirements and should not fall during a seismic event and local pipe whip restraints should prevent the monitors from becoming missiles after a postulated pipe break. Therefore, the proposed change will not create the possibility of a new or different kind of accident previously evaluated.

The monitor provides backup recording to verify seismic induced stress calculations. The new location still provides information on the RCS piping and the monitor should have no effect on the new piping location. The current location renders the monitor useless while the new location restores the margin of safety as a backup monitor as originally required. Therefore the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, the staff proposes to determine that the change does not involve a significant hazards consideration.

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

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

NRC Project Director: Jose A. Calvo

Mississippi Power & Light Company, System Energy Resources, Inc., South Mississippi Electric Power Association, Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: December 6, 1988, as supplemented December 30, 1988.

Description of amendment request: The amendment would change the Technical Specifications (TS) as required to support the Cycle 4 fuel reload. Specific changes would be made in the Bases for Section 2.1, "Safety Limits," the TS and Bases for Section 3/4.2, "Power Distribution Limits," and TS 5.3.1, "Fuel Assemblies." Specifically, the proposed Technical Specification changes address the following:

(a) The addition of one MAPLHGR curve for the new 8x8 fuel type.

(b) The revision of the MAPLHGR curve for 8x8 fuel during Single LOOP Operation (SLO).

(c) The revision of flow dependent thermal limits, $MAPFAC_p$, and $MCPR_f$, based on all ANF core for Cycle 4.

(d) The revision of power dependent MCPR, $MCPR_p$, based on analyses specific to an all ANF core for Cycle 4.

(e) Changes associated with the addition of four 9x9-5 Lead Test Assemblies (LTAs) introduced in Cycle 4. The applicable MAPLHGR and LHGR curves are added.

(f) The revision of design description of the fuel assemblies consistent with Item (e) above (administrative).

(g) Administrative changes (editorial).

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. a) This change introduces one MAPLHGR limit for the new 8x8 fuel. This change only introduces a new MAPLHGR limit and does not affect the precursors to any event previously evaluated. Therefore, this change does not involve a significant increase in the probability of any event previously evaluated.

The peak clad temperature (PCT) for the new 8x8 fuel was calculated based on the same bounding MAPLHGR limit which was used in the analyses for Cycles 2 and 3. [The MAPLHGR operating limits in the Technical Specifications for Cycles 2, 3 and 4 are bounded by the MAPLHGR limit used in LOCA analysis.] Small variations in PCT, compared to the bounding PCT calculated in Cycle 2, are observed as a result of minor fuel design differences (e.g., lattice radial enrichment, and therefore, power distribution). The maximum increase in PCT relative to Cycle 2 is 11 degrees F at 20 GWd/MTU. This increase is negligible compared to the calculated PCT which is more than 500 degrees below the 10 CFR 50.46 limit of 2200 degrees F. Therefore, the proposed change does not involve a significant increase in the consequences of any event previously evaluated.

b) This change consists of a revision to the SLO MAPLHGR limit for the 8x8 fuel types. It only redefines the SLO MAPLHGR limit and does not affect the precursors to any event previously evaluated. Therefore, this change does not involve a significant increase in the probability of any event previously evaluated.

The revised SLO MAPLHGR limit conservatively bounds, during Cycle 4, the individual MAPLHGR limits for the 8x8 fuel types. Therefore, this change does not involve a significant increase in the consequences of any event previously evaluated.

c) This change consists of revisions to the $MCPR_f$ and $MAPFAC_f$ limits. The revised limits are based on ANF's methodology, are defined for specific modes of operation and do not take credit for the core flow limiter. These changes only redefine the flow dependent thermal limits and do not affect the precursors to any event evaluated previously. Therefore, these changes do not involve a significant increase in the probability of any event evaluated previously.

As a result of this change, both reduction and increase in the Cycle 3 limits are observed. However, the revised $MCPR_f$ and $MAPFAC_f$ operating limits were constructed in a conservative manner. The limiting flow runout event will not cause the plant to exceed the MCPR safety limit or the LHGR 120% overpower line even with the plant initially at the revised operating limits. Therefore, the proposed changes do not involve a significant increase in the consequences of any event previously evaluated.

d) This change consists of a revision to the $MCPR_p$ limit. The revised $MCPR_p$ limit is based on ANF's methodology applied to a full ANF core. The limit is lower than the Cycle 3 limit above 40% of rated power up to, but not including, 100% power. Below 40% of rated power and at 100% power, the limit is unchanged. This change only redefines the $MCPR_p$ limit and does not affect the precursors to any event previously evaluated. Therefore, this change does not involve a significant increase in the probability of any event previously evaluated. Cycle 4 analysis demonstrated that the limiting events will result in a CPR above the $MCPR_p$ operating limit. Therefore, the proposed change does not involve a significant increase in the consequences of any event previously evaluated.

e) This change addresses the introduction of four (4) LTAs into the core for Cycle 4 operation. The thermal, mechanical and neutronic performance of the LTAs has been determined for the limiting events evaluated by ANF for Cycle 4. The LTAs have been determined for the limiting events evaluated by ANF for Cycle 4. The LTAs have been shown to be compatible with the co-resident 8x8 fuel assemblies. Therefore, introduction of the LTAs during Cycle 4 does not affect the precursors to any event evaluated previously for 8x8 fuel. Therefore, this change does not involve a significant increase in the probability of any event previously evaluated for 8x8 fuel. The Cycle 4 reload analysis shows that the LTA performance is bounded by the performance of the co-resident 8x8 fuel. This is ensured by the LTAs being placed in non-limiting core locations. Therefore, the introduction of LTAs does not involve a significant increase in the consequences of any event previously evaluated.

f) This change is administrative. Therefore, it does not involve a significant increase in the probability or consequences of an accident previously evaluated.

g) These changes are administrative. Therefore, they do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Overall, the proposed changes define parameters determined conservatively and consistent with the fuel which will be resident in the core during Cycle 4. They do not affect the precursors to any accident previously evaluated. These changes, therefore, do not involve a significant increase in the probability or consequence of any accident previously evaluated.

2. The new 8x8 fuel type is of a design similar to the fuel present in the core. It has been determined by ANF that the 9x9x-5 LTA is compatible with the 8x8 fuel and will not create the possibility of a new or different kind of accident. The proposed changes do not involve any new modes of operation, any plant modifications or any changes to setpoints. Therefore, the proposed changes do not result in the creation of any new precursors to any accident. They only introduce new and revised MAPLHGR, LHGR and off-rated power and flow limits. These limits have been determined using methodologies similar to those used for previous cycles. The administrative changes have no effect on any accidents. Therefore, the proposed changes do not create the possibility of a new or different type of accident from any accident previously evaluated.

3. a) This change introduces one MAPLHGR limit for the new 8x8 fuel. The peak clad temperature (PCT) for the new 8x8 fuel was calculated based on the same bounding MAPLHGR limit which was used in the analyses for Cycles 2 and 3. Small variations in PCT, compared to the bounding PCT calculated in Cycle 2 are observed as a result of minor fuel design differences (e.g., lattice radial enrichment, and therefore, power distribution). The maximum increase in PCT relative to Cycle 2 is 11 degrees F at 20 GWd/MTU. The available margin to the 10CFR50.46 limit of 2200 degrees F at this exposure is greater than 500 degrees F. Therefore, the introduction of the new MAPLHGR limit does not involve a significant reduction in the margin of safety.

b) This change consists of a revision to the SLO MAPLHGR limit for 8x8 fuel. The revised SLO MAPLHGR curve conservatively bounds, during Cycle 4, the individual MAPLHGR limits for all 8x8 fuel types. The method used to calculate off-rated MAPLHGR limits in Cycle 3 is maintained for Cycle 4. Therefore, revision of the SLO MAPLHGR limit does not involve a significant reduction in the margin of safety.

c) This change consists of revisions to the $MCPR_f$ and $MAPFAC_f$ operating limits. The revised limits are based on ANF's methodology, are defined for specific modes of operation and do not take credit for the use of the core flow limiter. The revised $MCPR_f$ limit is based on a conservative bound of the maximum achievable core flow (110% of rated) for the limiting flow runout event. The Cycle 3 limits are based on maximum core flows of 102.5 and 107% of rated. The revised $MCPR_f$ limits are in general lower than the Cycle 3 $MCPR_f$ operating limits. However, the ANF Cycle 4-specific safety analyses show an adequate margin to the safety limit. The $MCPR_f$ limit consists of two curves corresponding to Non-Loop Manual and Loop Manual modes of operation. For Non-Loop Manual modes, the limiting flow runout event consists of a two loop runout whereas for Loop Manual mode, the limiting event consists of a one loop runout. Therefore, the limiting consequences (flow increase and the associated delta CPR) in the Loop Manual mode are smaller than in the Non-Loop Manual modes, resulting in an added CPR margin for the LOOP Manual mode.

The $MCPR_f$ operating limit is constructed based on a number of conservative assumptions: 1) The increase in flow rate for both one and two loop runout events are [sic] conservative (see report NESDQ-88-003), 2) the ANF analysis assumes a conservative rod-line for the limiting flow runout event, and 3) the $MCPR_f$ limit includes an added conservatism to address performance variations in subsequent cycles (NESDQ-88-003 and ANF-88-149, Figure 5.1). With the plant initially at the revised $MCPR_f$ operating limit, the limiting flow run-out event, for both Loop Manual and Non-Loop Manual operations, will result in a final CPR above the MCPR safety limit. This ensures that an adequate margin of safety is available.

The basis for determining the $MAPFAC_f$ limits is similar to that used in determining the $MCPR_f$ limits. The $MAPFAC_f$ limit consists of two curves corresponding to Non-Loop Manual and Loop Manual modes of operation. The change in $MAPFAC_f$ under the more restrictive Loop Manual mode (one loop runout) is smaller than under the Non-Loop Manual modes (two loop runout). The conservatisms associated with the assumed flow increases and the analysis rod-line described above for the $MCPR_f$ related analyses are applied to the $MAPFAC_f$ analyses as well. With the plant initially on the revised $MAPFAC_f$ limit, the limiting flow runout event, for both Loop Manual and Non-Loop Manual operations, will result in a final $MAPFAC_f$ below the 120% overpower line. This assures an adequate margin of safety for this event.

Therefore, the proposed changes in the $MCPR_f$ and $MAPFAC_f$ limits do not involve a significant reduction in the margin of safety.

d) This change consists of a revision to the $MCPR_p$ limit. The revised $MCPR_p$ limit is based on ANF's methodology applied for a full ANF core. The limit is lower than the Cycle 3 limit above 40% of rated power up to, but not including, 100% power. Below 40% of rated power and at 100% power, the limit is unchanged. Cycle 4 analysis demonstrated that even with the plant initially on the revised $MCPR_p$ operating limit, the analyzed limiting core-wide transients and local events will result in a CPR above the MCPR safety limit. Therefore, the proposed change in the $MCPR_p$ limit does not involve a significant reduction in the margin of safety.

e) This change addresses the introduction of four (4) LTAs into the core for Cycle 4 operation. The thermal and mechanical performance of the LTAs for the limiting events analyzed by ANF for Cycle 4 is bounded by the performance of the 8x8 fuel. MAPLHGR and LHGR curves specific to the 9x9-5 LTA have been developed. These curves were developed using the same methods as were used for the 8x8 fuel. Comparable margins to the PCT and mechanical design limits were shown to be available for the LTAs. Additional margin is introduced by placing the LTAs in non-limiting core locations. Therefore, the introduction of four (4) LTAs does not involve a significant reduction in the margin of safety.

f) This change is administrative. Therefore, it does not involve a significant reduction in the margin of safety.

g) These changes are administrative. Therefore, they do not involve a significant reduction in the margin of safety.

Therefore, these changes ((a) through (g)) 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 Acting Project Director: Edward A. Reeves

Mississippi Power & Light Company, System Energy Resources, Inc., South Mississippi Electric Power Association, Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: December 16, 1988

Description of amendment request: The amendment would change the Technical Specifications (TS) Section 6.0, "Administrative Controls," by:

1. Replacing references to specific staff positions identified in the composition of the Plant Safety Review Committee (PSRC) and the Safety Review Committee (SRC) with descriptions and qualifications of required personnel.

2. Adding a footnote to TS Table 6.2.2-1, "Minimum Shift Crew Composition," to allow a licensed senior reactor operator (SRO) on the crew to serve in a dual capacity as SRO and shift technical advisor (STA).

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.

Change 1

No significant increase in the probability or consequences of an accident previously evaluated results from this change.

a. The replacement of specific position titles with general titles and requirements is administrative. The proposed change does not affect assumptions contained in plant safety analyses, the physical design or operation of the plant, nor are TS that preserve safety analysis assumptions affected. The same level of expertise applied to the PSRC and SRC review function will exist with the approval of the proposed change. There will be no loss in PSRC or SRC effectiveness due to the proposed change. The positions which are important to safe operation of the facility will continue to be specified in the TS. The NRC will continue to be informed of the PSRC/SRC composition through the UFSAR.

b. Therefore, there is no increase in the probability or consequences of previously analyzed accidents due to the proposed change.

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

a. The proposed change is administrative. No physical alterations of plant configuration or change to setpoints or operating parameters are proposed. The level of position qualifications are not reduced in the TS. The same level and quality of PSRC and SRC review is maintained and unaltered by this proposed change.

b. Therefore, the possibility of a new or different kind of accident from any previously evaluated is not created.

3. This change would not involve a significant reduction in the margin of safety.

a. The change being proposed is administrative and does not relate to or modify the safety margins defined in and maintained by the TS. The change does not alter SERI's commitment to maintain a management structure that contributes to the safe operation and maintenance of the plant. No position qualifications are being reduced in the TS. The level and quality of PSRC and SRC review is maintained since there will be no change in the collective talents on the PSRC and SRC the scope of independent review conducted by the PSRC and SRC will be unchanged.

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

Change 2

1. No significant increase in the probability or consequences of an accident previously evaluated results from this change.

a. The objective of the STA requirement is to improve the ability of an operating shift to recognize, diagnose and effectively deal with plant transients or other abnormal conditions. The analysis of accidents such as Rod Withdrawal Error, Rod Drop Accidents, etc., that concern operator error, do not take credit for the STA as decreasing the probability of occurrence of these accidents. The proposed change simply provides flexibility in meeting an administrative requirement and does not involve any modifications or change in the plant.

b. With the proposed change, GGNS operating shift personnel will continue to have the expertise to recognize and effectively deal with plant transients or other abnormal events. The analysis of accidents such as Rod Withdrawal Error, Rod Drop Accidents, etc., that concern operator error, do not take credit for the STA as mitigating the consequences of these accidents. Rather, these accidents are mitigated by plant design (i.e. Rod Pattern Control System, Shutdown Margin, Core Monitoring Instrumentation, etc.). The proposed change is administrative. The expertise of the operating shift is not jeopardized and the radiological consequences of any evaluated accident remain unchanged.

c. Therefore, there is no increase in the probability or consequences of previously analyzed accidents due to the proposed change.

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

a. The proposed change does not involve any modifications or changes in the plant. This is an administrative change in which the ability of the operating shift is not jeopardized. Since the STA has no operational responsibilities or duties on shift other than those associated with plant transients and accidents, combining the Shift Superintendent or the second SRO function with the STA will not introduce any new opportunity for operator error to occur.

b. Therefore, the possibility of a new or different kind of accident from any previously evaluated is not created.

3. This change would not involve a significant reduction in the margin of safety

a. The proposed change will not have any effect on safety limits, boundary performance or system performance. The STA or SRO/STA will continue to monitor thermal limits, thermal power, core flow, reactor pressure and level to ensure safety limits are not exceeded in normal or abnormal situations.

b. The functions of the STA will continue to be carried out by an individual on shift. That individual on shift will continue to have the knowledge, training, experience, and expertise required to assess, analyze, and evaluate plant transients and accidents. There will be no detraction from the operating duties of the SRO or STA.

c. The proposed change still would meet the current NRC position on training and qualification of STAs. In addition, NRC shift staffing requirements would still be met with the proposed change.

d. Therefore, this proposed change will not involve a 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 proposed 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, 1400 L Street, NW., Washington, DC 20005

NRC Project Director: Elinor G. Adensam

Mississippi Power & Light Company, System Energy Resources, Inc., South Mississippi Electric Power Association, Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: January 26, 1989

Description of amendment request: The amendment would provide one-time exceptions to TS 3.0.4 for certain Technical Specifications (TS) during the third refueling outage while the plant is in Operational Condition 4 (cold shutdown) and Operational Condition 5 (refueling). For these TS, the exceptions to TS 3.0.4 would allow entry into the specified operational conditions without meeting limiting conditions for operation provided the requirements of the associated action statements are met. The use of these exceptions will reduce the refueling outage time. The specific TS for which exceptions to TS 3.0.4 are requested are:

a. Residual Heat Removal - Cold Shutdown, 3.4.9.2, Actions a and - page 3/4 4-27

b. ECCS - Shutdown, 3.5.2, Action a - page 3/4 5-6

c. Suppression Pool, 3.5.3, Action c - page 3/4 5-9

d. Containment and Drywell Isolation Valves, 3.6.4, Actions b and c - page 3/4 6-28

e. Secondary Containment Automatic Isolation Dampers/Valves, 3.6.6.2, Actions b and c - page 3/4 6-49

f. Standby Service Water System, 3.7.1.1, Actions b, c and d - pages 3/4 7-1 and 3/4 7-2

g. Ultimate Heat Sink, 3.7.1.3, Action a - page 3/4 7-4

h. Control Room Emergency Filtration System, 3.7.2, Action b.1 - page 3/4 7-5

i. Residual Heat Removal and Coolant Circulation - Low Water, 3.9.11.2, Actions a and b - page 3/4 9-19

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. The proposed changes are intended to provide operational flexibility during the upcoming refueling outage while ensuring core decay heat removal capability, ECCS water injection requirements and primary and secondary containment capability. SERI has developed and implemented a management philosophy for effective control of potential vessel draining and decay heat removal during plant outages. This philosophy has been implemented by policy as a Technical Specification Position Statement which requires:

a) At least one ECCS and one Fuel Pool Cooling subsystem functional at all time.

b) At least one shutdown cooling subsystem of RHR remain functional except for periods of required maintenance or testing.

c) The emergency diesel/generator associated with the one required ECCS, Fuel Pool Cooling, and Shutdown Cooling subsystem be functional (and OPERABLE when possible).

d) Any alternate shutdown cooling subsystem must be demonstrated to be able to remove reactor decay heat load existing at the time the system is required.

In addition, it is SERI's outage philosophy to minimize the time in TS action statements associated with the above systems such that these action statements are only entered for required maintenance, testing, inspections, and modifications. Any exceptions to the above must receive prior Plant Safety Review Committee review and approval.

2. This policy has been successfully executed and demonstrated effective in previous refueling outages.

3. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. SERI has evaluated UFSAR Chapter 15 events which are considered to be applicable during OPERATIONAL CONDITIONs 4 and 5. These events include a dropped fuel bundle and inadvertent criticality. The proposed Specification 3.0.4 exceptions cannot affect the probability of occurrence of any of these events. The proposed 3.0.4 exceptions would have no effect on fuel handling operations in the containment or in the spent fuel pool because fuel handling procedures and methods remain unchanged. The proposed changes have no effect on control rod interlocks or fuel loading errors and thus do not affect the probability of occurrence of an inadvertent criticality. The proposed changes will allow the following evolutions to occur during the third refueling outage while in the action statements of the affected TS:

a. Tensioning and detensioning the reactor vessel head.

b. Lowering the reactor cavity water level to less than 22 feet 8 inches above the reactor pressure vessel flange.

c. Performance core alternations and handling irradiated fuel while relying on the provisions of ACTION b and c of TS 3.6.4 and 3.6.6.2.

4. The above listed evolutions will be performed while in the action statements associated with ECCS operating and shutdown requirements, provisions concerning the number of RHR shutdown cooling loops required OPERABLE, provisions concerning primary containment, drywell and secondary containment capability and control room emergency filtration system. Without the requested TS 3.0.4 exceptions, the required systems would have to be made operable just to perform the above evolutions and then they may be made inoperable again for maintenance and testing purposes. The evolution of making systems operable just to change operational conditions or other specified conditions represents significant impact on the refueling outage. With the proposed changes the outage length can be significantly decreased with no significant impact to overall plant safety.

5. The proposed changes do not affect the consequences of an accident previously evaluated. SERI policy looks at the overall outage plan and attempts to optimize testing and maintenance periods on ECCS and decay heat removal systems in order to ensure optimum availability while at the same time accomplishing required maintenance and testing activities.

6. The proposed changes involving RHR shutdown cooling affect Specifications 3.4.9.2 ACTIONs a and b, 3.7.1.1 ACTIONs b and d, 3.7.1.3 ACTION a, and 3.9.11.2. The action statements of Specifications 3.4.9.2 and 3.9.11.2 contain provisions to establish alternate methods of decay heat removal, when necessary, with RHR shutdown cooling loops inoperable. These alternate methods of decay heat removal are procedurally prescribed prior to entering an outage based on available equipment and planned outage activities. Since decay heat removal is provided for in the action statements of the affected specifications, entry into the OPERATIONAL CONDITIONs with less than the required number of RHR Shutdown Cooling Loops available does not involve a significant increase in the probability or consequences of an accident previously evaluated.

7. The proposed change to Specification 3.7.1.1 ACTIONs b and and d and 3.7.1.3

ACTION a affect the SSW subsystems and ultimate heat sink that support the RHR shutdown cooling loops. With an SSW subsystem inoperable, its associated RHR shutdown cooling loop is also required by Technical Specifications to be declared inoperable. Changing OPERATIONAL CONDITIONs or other specified conditions with this SSW subsystem and associated RHR shutdown cooling loop inoperable represents no significant increase in the probability or consequences of an accident previously evaluated.

8. The proposed changes to Specification 3.5.2 and 3.7.1.1 ACTION c will allow operational condition changes with one ECCS subsystem/system OPERABLE. Since only OPERATIONAL CONDITIONs 4 and 5* are affected, present TS indicate that one ECCS subsystem/system is sufficient for water makeup requirements for the four hour time allowance of ACTION c of Specification 3.5.2. The proposed change to ACTION c of Specification 3.7.1.1 is similar to that for ACTIONs b and d such that when equipment is out of service, a support system such as SSW is not required to be OPERABLE for that ECCS function. Since ECCS makeup capability is provided while in ACTION a of Specification 3.5.2, the proposed change does not involve a significant increase in the probability of consequences of an accident previously evaluated.

9. The proposed change to 3.5.3 ACTION c will allow operation of the Alternate Decay Heat Removal System (ADHRS) which requires declaring inoperable a division of suppression pool water level instrumentation. TS 3.0.4 presently restricts changing operational conditions while relying on the provisions of that action. ADHRS operation causes the inoperability of one division of suppression pool level instrumentation which causes entry into ACTION c of TS 3.5.3. This action requires that suppression pool level be verified once per 12 hours by an alternate indicator. Operational condition or specified condition changes cannot be made while relying on the provisions of the ACTION even though suppression pool level can be verified by an alternate indicator. Since an alternate means of verifying suppression pool level is provided by ACTION c of Specification 3.5.3, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

10. The proposed changes involving drywell, primary containment and secondary containment isolation valves affect Specification 3.6.4 ACTIONs b and c and Specification 3.6.6.2 ACTIONs b and c. The action statements of those specifications provide provisions for isolating affected penetrations when one or more of the associated isolation valves or dampers are inoperable. The action involves isolating the affected penetration by use of at least one deactivated automatic valve secured in the isolated position or by use of at least one closed manual valve or blind flange such that the safety function of the valve or damper is accomplished. Because the affected penetrations are isolated in accordance with the specified actions, changing operational or other specified conditions while relying on the provisions of the action does not involve a significant increase in the probability or consequences of an accident previously evaluated.

11. The proposed change involving the control room emergency filtration system affects Specification 3.7.2 ACTION b.1. The action statement of that specification provides provisions for OPERATIONAL CONDITIONs 4, 5 and "*" when one of the two required control room emergency filtration system subsystems are inoperable. The action requires restoration of the inoperable subsystem within seven days or initiate and maintain operation of the OPERABLE subsystem in the isolation mode of operation. Since emergency filtration capability is provided by the OPERABLE subsystem, changing operational conditions or other specified conditions with less than the required number of control room emergency filtration subsystems does not involve a significant increase in the probability or consequences of an accident.

12. The proposed change does not increase the possibility of a new or different kind of accident from any previously analyzed. The proposed changes do not increase the amount of time ECCS, RHR shutdown cooling loops or control room emergency filtration subsystems are unavailable nor do the changes reduce the drywell, containment or secondary containment isolation capability. The proposed changes do not increase the potential for draining the reactor vessel. Since the above safety systems are maintained, there is no possibility of a new or different kind of accident from any previously analyzed. The proposed changes are intended to increase outage flexibility while maintaining necessary levels of plant safety.

13. The proposed change does not involve a significant reduction in a margin of safety. The proposed Specification 3.0.4 exceptions will still ensure that core decay heat removal, ECCS makeup capabilities, control room emergency filtration capability, and drywell, containment and secondary containment capability are available when required during the refueling outage. In addition to Technical Specification action requirements, SERI is to maintain at least one ECCS system and one Fuel Pool Cooling and Cleanup system functional at all time during the outage. RHR shutdown cooling loops will be functional unless maintenance or testing removes them from service. SERI's outage policy will minimize time in the action statements as much as possible. Since essential safety systems are available as necessary during the outage, the change does not involve a significant reduction in a 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 for operational condition 4 and 5 only. 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: Edward A. Reeves

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

Date of amendment request: October 14, 1986, July 21, 1987 and January 12, 1989.

Description of amendment request: By applications for license amendments dated October 14, 1986, July 21, 1987 and January 12, 1989, Northeast Nuclear Energy Company (the licensee) requested changes to the Technical Specifications (TS) for Millstone Unit 2 to address recommendations of Generic Letter 83-37. The proposed change to the TS would incorporate Limiting Conditions for Operation (LCO) and Surveillance Requirements (SRs) for the Reactor Vessel Coolant Level (RVCL) instrumentation into TS 3/4.3.3.8, "Instrumentation - Accident Monitoring."

Basis for proposed no significant hazards consideration determination: The RVCL instrumentation for Millstone Unit 2 is based upon the heated junction thermocouple technology for post-accident determination of reactor pressure vessel water inventory. In our safety evaluations dated April 18, 1985 and August 28, 1986, the NRC staff addressed the adequacy of the RVCL instrumentation for Millstone Unit 2. The need for RVCL instrumentation and associated TS was one of a number of post-TMI initiatives that had been established by the NRC staff. Based upon discussions with the NRC staff, and applications for license amendments dated October 14, 1986 and July 21, 1987, the licensee has submitted revised proposed LCOs and SRs for the RVCL instrumentation in a letter dated January 12, 1989.

The proposed LCO for the RVCL instrumentation would require at least one of the two channels to be operable. In the event that no channel is operable either restore the unoperable channel(s) to operable status in 48 hours or:

1. Prepare and submit a special report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action

taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status; and

2. Restore the system to operable status at the next scheduled refueling; and

3. Initiate an alternate method of monitoring the reactor vessel inventory.

The SRs for the RVCL instrumentation includes monthly channel checks (a determination of operability) and calibration of the instrumentation (from the electronic cabinets only) during refueling. The approval of similar, generic, requirements is contained in a letter from Mr. D. Crutchfield, NRC, to Mr. R.W. Wells, Chairman, Combustion Engineering Owners Group, dated October 28, 1986.

On March 6, 1986, the NRC provided guidance in the **Federal Register** (51 FR 7751) concerning examples of amendments that are not likely to involve significant hazards consideration. One example of amendments not likely to involve significant hazards considerations is example (ii) which involves "A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications, e.g., a more stringent surveillance requirement." The proposed change to TS 3/4.3.3.8 would incorporate LCOs and SRs for the RVCL instrumentation into the TS. The proposed change to the TS is thus judged to be within the scope of example (ii), above. Accordingly, the Commission proposes to determine that the proposed change to the TS involves no significant hazards considerations.

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

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

NRC Project Director: John F. Stolz

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: January 6, 1989 as supplemented by letter dated January 20, 1989.

Description of amendment request: By application for license amendment dated January 6, 1989 as supplemented by letter dated January 20, 1989, Northeast Nuclear Energy Company, et al. (the licensee), requested changes to Millstone Unit 3 Technical Specification (TS) 4.7.10b, "Snubbers", to allow an approximate two month extension in snubber visual inspections, to allow continued operation until the next refueling outage.

Technical Specification 4.7.10b requires that snubbers on safety-related components and piping be visually inspected at various intervals depending upon snubber failure rate determined by the previous inspection. An increased number of snubber failures would decrease the surveillance intervals from as great as 18 months 27 25% to as little as 31 days 27 25%. The current inspection interval for Millstone Unit 3 is 18 months for all snubbers except for Type PSA-1/2 and PSA-1/4, which have a 12 month interval. During the last round of inspections, the licensee found all snubbers operable which enabled the licensee to increase the inspection interval for the Type PSA-1/2 and PSA-1/4 snubbers to 12 months. The next required inspection interval would end April 30, 1989. The licensee has requested that the surveillance interval be extended to allow snubber inspection during the next refueling outage, which is scheduled to begin on May 20, 1989.

Basis for proposed no significant hazards consideration determination:

Title 10 CFR 50.92, "Issuance of amendment", contains standards for evaluating the existence of no significant hazards consideration. In this regard, the proposed change to TS 4.7.b will not:

- Involve a significant increase in the probability or consequences of an accident previously evaluated. The probability of a seismic event is independent of the snubber surveillance program. With regard to consequences of seismic events, it is unlikely that a one time extension of approximately 20% of the snubber inspection interval will appreciably increase the incidence of undetected snubber failure. The inherent seismic-resistance capability of the components and piping provide reasonable assurance of safety during the proposed extended inspection interval.
- Create the possibility of a new or different kind of accident. Safety systems that were designed to be seismic-resistant, will continue to be seismic-resistant with no significant decrease in capability. Thus, no new or different types of accidents will be created as a result of seismic events.
- Involve a significant reduction in a margin of safety. Although there may be small, localized, reductions in safety margins with regard to seismic resistance of safety systems due to undetected snubber failures, the overall reduction in safety margin will not be significant. The proposed change does not affect the consequences of any accident previously analyzed.

Based on the above, the staff proposes to determine that the proposed change to TS 4.7.10.b does not involve a 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

Northern States Power Company, Dockets Nos. 50-282 and 50-306, Prairie Island Nuclear Generating Plant, Units Nos. 1 and 2, Goodhue County, Minnesota

Date of amendment request: July 18, 1988.

Description of amendment request: The proposed Technical Specification (TS) changes would eliminate requirements dealing with steam generator low water level and low feedwater flow. Specifically, the proposed changes to the TSs, which would become effective after the installation of the digital feedwater control system, are as follows:

1. Specification 2.3.A.3(c) dealing with the reactor trip setpoints of "low steam generator water level - greater than or equal to 15% of the narrow range instrument in coincidence with steam/feedwater mismatch flow - greater than or equal to $1.0x10^6$ lbs/hr" would be deleted.

2. Specification Table TS.3.5-2, item 18 dealing with low feedwater flow reactor trip, would be deleted.

3. Specification Table TS.4.1-1, item 12, Steam Generator Flow Mismatch, would be modified so that surveillance would be performed on steam flow channels only since feedwater flow channels would no longer be used in the protection circuit.

The licensee also proposes to revise the bases to reflect the removal of the low feedwater flow reactor trip.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of standards for making a no significant hazards consideration determination by providing certain examples (51 FR 7751). One of the examples is (ix):

A repair or replacement of a major component or system important to safety, if the following conditions are met:

(1) The repair or replacement process involves practices which have been successfully implemented at least once on similar components or systems elsewhere in the nuclear industry or in other industries, and does not involve a

significant increase in the probability or consequences of an accident previously evaluated or create the possibility of a new or different kind of accident from any accident previously evaluated; and

(2) The repaired or replacement component or system does not result in a significant change in its safety function or a significant reduction in any safety limit (or limiting condition of operation) associated with the component or system.

The replacement feedwater control system that utilizes a median signal selector function has been installed at several other plants where it demonstrated a superior means of feedwater flow control as compared to the existing control systems. This advanced means of controlling feedwater flow eliminates the possibility of flow transient conditions, and therefore the need for a reactor trip initiated by low feedwater flow or low steam generator water level becomes unnecessary. The setpoint parameters associated with the steam generator water level and feedwater flow have not been factored into analyses of any of the previously analyzed accidents. Therefore, the elimination of these reactor trip settings does not involve a significant increase in the probability or consequences of an accident previously evaluated or create the possibility of a new or different kind of accident from any accident previously evaluated. In addition, the proposed changes will in no way alter the safety function of the feedwater control system or result in a significant reduction in any safety limits associated with the feedwater control system. On this basis, the Commission proposes to determine that the requested action does not involve a significant hazards consideration.

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

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

NRC Project Director: Theodore R. Quay, Acting.

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

Date of application for amendments: January 26, 1989

Description of amendment request: The proposed amendment would modify the Technical Specifications to correct deficiencies in the degraded voltage protection features. The amendment replaces in its entirety an earlier amendment dated September 7, 1988 for which notice of consideration was provided in the **Federal Register** on October 19, 1988 [53 FR 40996]. Accordingly, this notice replaces and supersedes in its entirety the Notice of October 19, 1988. The deficiencies were identified as a result of revised voltage regulation studies. The studies were based in part on the consideration that, under certain offsite power emergency conditions, the voltage provided to the station's offsite power supply transformers could be lower than previously assumed. The study also modeled the plant's power distribution system to a greater level of detail.

The proposed changes are grouped into two categories. The Category A changes address the degraded grid protection relays, and involve providing protective relays on each 4.16kV bus (with revised voltage setpoints) and increasing the time delay for the 4.16kV bus to transfer to an alternate power supply. Category B changes address the Emergency Core Cooling System (ECCS) loading sequence.

The Category A changes involve two independent offsite power sources which are referred to as the start-up sources. The 4160 volt (4.16kV) bus feeder breakers provide the interface between the two offsite power sources and the plant safety-related AC power distribution system. Each of the four 4.16kV buses in each unit can be powered by either of the two offsite power supplies. Each of the 4.16kV buses can also be powered from a safety related diesel generator.

Each startup source to each 4.16kV bus is equipped with an instantaneous undervoltage protective relay. Each relay is presently set to initiate at 90% of nominal voltage on the 4.16kV bus. The purpose of these relays is to ensure that adequate levels of voltage are provided to the motors and control components which are powered from the 480V motor control centers (MCCs) which are fed from the 4.16kV buses. After a 0.1 second internal time delay the degraded voltage protective relays initiate time delay relays which transfer the 4.16kV bus to an alternate supply source if the normal supply source does not recover to the instantaneous relay reset value (currently 93%) in a set period of time. The control circuit logic to the time delay relays distinguishes between an undervoltage condition without a safety injection signal and one concurrent with a safety injection signal. Without a safety injection signal, a time delay relay will initiate the transfer 60 seconds after initiation of the instantaneous relay if the voltage does not recover.

With a safety injection signal, another time delay relay will initiate the transfer six seconds after initiation of the instantaneous relay if the voltage does not recover. The purpose of the six second delay is to minimize the time that safety-related equipment is exposed to the undervoltage condition, yet allow the voltage to recover from the dips caused by acceleration of the large safety-related motors. In either case, if the voltage of the normal supply has not recovered before the time delay relays initiate the transfer, the associated source breaker is tripped and the bus is loaded onto an alternate power supply. The alternate supply for an 4.16kV bus is, in order of preference, the remaining offsite power source, then the emergency diesel generator. The revised voltage regulation study identified that under the scenario of a safety injection signal on one unit while operating with only one of two offsite power sources (permitted for seven days by Limiting Condition for Operation 3.9.B.1), the existing six seconds time delay setting is inadequate. The existing six second timer setting, along with the 0.1 second internal delay, would not allow sufficient acceleration time for the core spray pump motors. Therefore, even after a 6.1 second delay, the core spray pump motors, which are currently started simultaneously, will not be at rated speed (based on design acceleration versus voltage values) thereby not allowing voltage recovery on the 4.16kV buses, and all four 4.16kV bus feeder breakers will trip, thus loading each bus onto its associated diesel generator. This would represent a reduction in defense in depth since it is desirable, if offsite power is available, to supply these loads from the offsite power supply without reliance on the backup diesel generators. The licensee has identified two categories of changes to address this concern.

The Category A changes deal with the offsite power source and include the following: (1) Revise Technical Specification Table 3.2.B on page 71a to designate the presence of undervoltage protective relays (two per 4.16kV bus) which actuate under LOCA conditions and set at "89% of rated voltage 27 0.3% of setting (3702 volts 27 11 volts) with a "0.9 - 1.1 second internal time delay" and undervoltage protective relays (two per 4.16kV bus) which actuate under non-LOCA conditions and are set at "98% of rated voltage 27 0.3% of setting (4077 volts 27 12 volts)" with a "0.9 - 1.1 second internal time delay" instead of

"90% (+/-2%) of rated voltage," and replace the "(ITE)" in the trip function column with "(27N)"; (2) Revise Table 3.2.B on page 71a to designate the trip level setting for the LOCA time delay relays as "9 second 27 7% (27 0.6 sec.) time delay" instead of "6 second (+/-5%) time delay." Express the tolerance of the "non-LOCA" relay in terms of seconds (27 5% as 27 3 sec.); (3) Revises Bases section 3.2 on page 93a to reflect the presence of separate relays for LOCA and non-LOCA conditions, with the LOCA relay set at 89% and the non-LOCA relay set at 98%.

The Category B changes deal with revising the scheme for the sequential loading of the residual heat removal (RHR) and the core spray (CS) pumps. The four CS pumps and the four RHR pumps of the Emergency Core Cooling System (ECCS) are powered from the 4.16kV buses. In the event of a LOCA with offsite power available, the RHR and CS pumps are loaded sequentially onto the 4.16kV buses to preclude severe voltage transients from the simultaneous starting of the pumps. The present loading sequence for the RHR and CS pumps in the event of the safety injection signal with offsite power available results in voltage dips on the 4.16kV and 480V buses which are unacceptable in consideration of the degraded grid protective relay settings due to core spray pump motor acceleration time. Therefore, the licensee proposes a revised loading sequence for a safety injection signal with offsite power available as follows: (1) Revise Table 3.2.B on page 67 to designate the initiation setpoint for the A and C core spray pumps to be "13 sec. +/-7% of setting" and the initiation setpoint for the B and D core spray pumps to be "23 sec. +/-7% of setting"; (2) Revise Table 3.2.B on page 67 to designate the initiation setpoint for the A and B LPCI pumps to be "2 sec. +/-7% of setting" and the initiation setpoint for the C and D LPCI pumps to be "8 sec. +/-7% of setting"; (3) Revise Table 3.2.B on page 67 of the Unit 3 Technical Specifications only to delete the asterisk next to the ADS Bypass Timer and the footnote which reads "Effective when modification association with this amendment is complete."

In addition to the proposed ECCS loading sequence, the licensee will further improve the voltage regulation of the 480V load centers during a motor starting transient by a combination of plant modifications which revise the load shedding or sequencing of the emergency service water pumps, the emergency cooling water pump, the RHR compartment coolers, the cooling towers and the diesel generator vent supply fans. The licensee plans to perform these changes pursuant to 10 CFR 50.59 since none involves an unreviewed safety question or a change to the Technical Specifications. The Appendix K (ECCS Evaluation Models) analysis was used to determine bounding allowable starting times for the RHR and CS pumps. For change Request (1), the licensee concluded that the proposed increases in the core spray timer settings are within the Appendix K analysis. Success of the core spray system requires two factors: (1) pump ready for rated flow and (2) injection valve open to permit full flow. There are two conditions required to support worst case valve opening; reactor pressure is at the low end of its low pressure permissive (400-500 psig) and power is available to the valve operator. Under the limiting scenario, the low pressure permissive occurs 47 seconds following occurrence of the LOCA. Power to the injection valves is not interrupted in this scenario and the valve stroke time is 12 seconds. The earliest that the injection valve can be opened, therefore, is 59 seconds, and the pumps must be ready for full flow prior to this time. The series of events contribution to the establishment of the pumps ready for rated flow are the sensor times for detection of the LOCA, the time for power to be available at the emergency bus, the time for power to be available to the pump motor and pump motor acceleration time. As stated previously, an assumption of the current Appendix K analysis of record is that the time available to start and accelerate the CS pumps from the offsite sources is 59 seconds. Taking into account the above equipment operational time requirements, the CS timer setting must be less than 47 seconds. Thus, the proposed 13 and 23 second timer settings are within the analyzed condition.

For Change Request (2), the licensee has similarly concluded that the proposed increases in RHR pump timer settings are in accordance with the Appendix K analysis. Success of the low pressure coolant injection (LPCI) mode of the RHR system requires three factors: (1) pump ready for rated flow, (2) injection valve open to permit full flow and (3) full closure of the recirculation discharge valve. Under the limiting scenario, 57 seconds are available for the RHR pumps to start and accelerate to rated speed. The 57 seconds are derived from the time to reach the low pressure permissive to close the reactor recirculation discharge valve plus the full stroke closure time of the recirculation discharge valve. The series of events for the RHR pumps ready for rated flow are similar to the series of events for the CS pumps. Taking into account the sensor and acceleration delays, the RHR timer setting must be less than 50.9 seconds. Thus, the proposed two and eight second timer settings are within the analyzed condition. Neither change request involves additional loading onto the DC system. All replacement and additional relays resulting from these changes will be located in existing safety-related panels. The control relays provided will equal or exceed the ratings of the existing relays and meet the applicable design requirements for environmental and seismic qualification.

Change Request (3) is proposed to the Unit 3 Technical Specifications only to delete a footnote which is no longer required since the modification associated with the ADS bypass timer (Modification 633) was completed for Unit 3 on February 24, 1986. Removing the footnote will eliminate the need to check the status of the modification to determine the applicability of the specification. The licensee proposes this administrative change to enhance safety by reducing the effort required to interpret the specification.

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 considerations 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; (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 a discussion of the proposed changes as they relate to these standards; the discussion is presented below.

Standard 1 - The proposed Category A changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Category A changes are proposed to improve the protection provided by the undervoltage protective relays. The application of two undervoltage relays per the proposed logic scheme represents a significant improvement in the level of protection provided to 480 volt MCC components under normal (non-LOCA) conditions. Although the proposed setpoint for the undervoltage relay used for protection in the event of a LOCA is lower than the existing relay setpoint, protection to the MCC components is actually improved due to the improved operational tolerance of the proposed replacement relay. Increasing the setting on the "LOCA" time delay relay from 6 seconds to 9 seconds will ensure that he 4.16kV buses will not be spuriously transferred to the diesel generators in the event of a design basis accident with only one offsite power source available. These proposed changes do not affect the probability or consequences of any accidents previously evaluated, but ensure that the 4.16kV buses will not be spuriously transferred to the diesel generators thereby ensuring the validity of the existing accident analysis; specifically, a loss of coolant accident with off-site power available.

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

The proposed changes to the relay settings do not involve a redistribution of loads on safety-related buses or affect the electrical separation or redundancy of any safety-related trains or components. The proposed changes improve the undervoltage protective scheme and allow the 4.16kV buses to sustain a normal motor acceleration transient without a spurious transfer to an alternate power source. The Category A changes do not alter the intent of the relays, and do not create the possibility of a new or different kind of accident from any previously evaluated.

Standard 3 - The proposed Category A changes do not result in a significant reduction in a margin of safety.

The Category A changes are proposed to enhance safety. The proposed change in undervoltage protection results in an improved protected voltage level to 480 volt MCC's and associated control components for both LOCA and non-LOCA conditions. The tolerance for the existing undervoltage relays is 272% of setting. The tolerance for the proposed undervoltage relays is 270.3% of setting. This results in an improved minimum protected level for non-LOCA conditions from 88.2% of rated voltage to 97.7% of rated voltage, and an improved minimum protected level for LOCA conditions from 88.2% of rated voltage to 88.7% of rated voltage.

The "non-LOCA" setpoint assures a limiting voltage value of 93% to 480 volt MCC's. An associated review of MCC contactor control circuits and implementation of control circuit modifications as necessary will assure 85% voltage to contactors. The "LOCA" undervoltage relay comes into effect on a LOCA signal, and the "non-LOCA" or normal protective setpoint is inhibited on the LOCA signal. The transition between the "non-LOCA" and "LOCA" undervoltage relays in essence represents a continuity of protection with respect to the offsite power sources to the 4.16kV buses when the effect of starting the 4kV ECCS motors on the 4.16kV buses is considered. Thus an improved continuity of protection against negative consequences of degrading grid or failure of offsite power source equipment is assured.

Increasing the time delay settings allows pump motors to accelerate without an unnecessary transfer to an alternate power supply. The changes do not involve a significant reduction in any margin of safety.

Standard 1 - The proposed Category B changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

The Category B changes are proposed to ensure the validity of the existing accident analyses; specifically, a design basis LOCA with offsite power available. Revising the timer settings for the RHR and CS pumps will improve the voltage at the 480V levels during a motor acceleration transient and also prevents spurious transfer of the 4.16kV buses to the diesel generators in the event of a safety injection while operating with only one offsite power source available. Therefore, the proposed changes do not increase the probability or consequences of an accident previously evaluated.

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

The proposed changes to the CS and RHR systems only involve changes to load sequencing when offsite power is available. The proposed changes do not involve the CS or RHR system piping configurations, pumps, valves or system redundancies. The replacement timers required for the proposed load sequencing equal or exceed the ratings for the existing timers, and do not affect the environmental or seismic qualification of the panels in which they will be installed. Failure of any timer can only affect one redundant train of equipment. Therefore, the possibility of a new or different kind of accident is not created.

Standard 3 - The proposed Category B changes do not result in a significant reduction in a margin of safety.

The proposed changes do not adversely affect the safety margin assumed in the 10 CFR Appendix K analysis for ensuring fuel integrity for the entire spectrum of postulated LOCAs. The limiting Appendix K scenario for core spray requires the CS pumps to be at rated flow 59 seconds after a LOCA to ensure the existing margin of safety. Under the proposed changes, the latest that the CS pumps will achieve rated flow is 35 seconds (3 seconds for detection of the LOCA plus 23 seconds for the longer of the CS timer delays plus a maximum of 9 seconds for motor acceleration). The limiting Appendix K scenario for the low pressure coolant injection mode of residual heat removal requires the RHR pumps to be at rated flow 57 seconds after a LOCA to ensure the existing margin of safety. Under the proposed changes, the latest that the RHR pumps will achieve rated flow is 14.1 seconds (3 seconds for detection of the LOCA plus 8 seconds for the longer of the RHR timer delays plus 3.1 seconds for motor acceleration). Therefore, although the Category B changes delay the availability of the CS and RHR pumps at rated flow, they do not result in a significant reduction in the margin of safety for core coolant delivery.

The staff has reviewed the licensees' no significant hazards consideration for Category A, items 1 and 2 and Category B, items 1 and 2 and agrees with the licensees' analysis. Accordingly, the Commission has proposed to determine that the above changes do not involve a significant hazards consideration.

The Category B, item 3, changes involving deletion of a now obsolete footnote is proposed as an administrative change to improve the use of the Technical Specifications. The Commission has provided guidance for the application of the criteria for no significant hazards consideration determination by providing examples of amendments that are considered not likely to involve significant hazards considerations [51 FR 7751]. These examples include: Example (i) "A purely administrative change to technical specifications: for example, a change to achieve consistency throughout the technical specifications, corrections of an error, or a change in nomenclature." The proposed change, to delete a footnote which refers to a now completed modification is an example of such an administrative change since, now that the modification has been completed, the specification is in effect and the footnote is extraneous. Since this proposed change is encompassed by an example for which no significant hazard exists, the staff has made a proposed determination that it involves no significant hazards consideration.

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

Attorney for Licensee: Troy B. Conner, Jr., 1747 Pennsylvania Avenue, NW. Washington, DC 20006

NRC Project Director: Walter R. Butler

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: December 2, 1988

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

Using the guidance provided by Generic Letter 87-09, this proposed change will clarify applicability of limiting conditions for operation and associated action requirements when a surveillance requirement is not performed within its allowed surveillance interval. It will state that a missed surveillance shall constitute noncompliance with the operability requirements of the related LCOs. It will specify that time limits for required actions for operating in a degraded mode apply at the time it is identified that a surveillance requirement has not been performed.

For allowable outage times that are less than 24 hours, a 24 hour delay period will be added to allow performance of a missed surveillance to satisfy operability requirements before implementing action requirements applicable to operating in a degraded mode.

The basis will be expanded accordingly to ensure the proposed changes for missed surveillance requirements are implemented consistent with the guidance provided in GL 87-09.

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 analysis:

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

Response:

A significant increase in the probability or consequences of an accident previously evaluated is not involved. A small increase in risk is associated with delaying the implementation of an LCO for 25 hours to allow completion of a missed surveillance. This risk is offset by a reduction in the possibility of a plant upset and challenge to safety systems. The risk of plant upset is greater if testing to complete a surveillance requirement is in progress at the time plant shutdown is commenced to comply with an LCO. It is preferable to allow time to complete the surveillance and demonstrate operability prior to changing plant status. The increase in safety gained from demonstrating operability during the delay period balances out the risk associated with the delay. In the case where inoperability is determined by testing during this extension, plant safety is enhanced if the affected equipment can be restored to an operable status prior to changing the plant's operating condition.

(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, as analyzed, does not involve a new or different kind of accident, from that previously evaluated. The definition of operability is clarified for the case of a missed surveillance. The application of LCO action requirements is expanded upon in this case and a delay is allowed by this proposed change to complete a missed surveillance before taking required actions. This affects only the impact of surveillance activities on plant operations by providing interpretation to the operator regarding the implementation of associated LCOs. Therefore, the possibility of a new or different kind of accident is not created.

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

Response:

A significant reduction in a margin of safety is not involved. An allowance for testing while operating is incorporated in the design of safety systems provided to prevent plant transients from approaching margins of safety. By allowing the completion of a missed surveillance before applying LCO shutdown requirements, this change will in fact reduce the potential for a challenge to safety systems while they are undergoing required testing.

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, Director

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: December 30, 1988.

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

This application seeks to amend Section 3.3 and Section 4.4 of Appendix A to the Operating License by revising the Limiting Condition for Operation (LCO) for the Weld Channel and Penetration Pressurization System (WC&PPS) and the Isolation Valve Seal Water System (IVSWS) to more closely reflect the system design. The proposed LCO changes will apply to the four independent zones of the WC&PPS and the individual station headers of the IVSWS, rather than to the supply headers of these systems. Consistent with the Westinghouse Standard Technical Specifications the allowable out-of-service time for one individual zone or station header of these systems will be seven days. The proposed change will also relocate an LCO from the Surveillance Requirements, Section 4.4 to Section 3.3.

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 analysis:

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

Response

The proposed change involves a revision in application of the WC&PPS and IVSWS operability requirements to more closely reflect system design and safety function. As the safety function and operability requirement of the WC&PPS is to provide compressed air to containment penetrations and liner weld channels, the LCO is clarified to specifically apply to those system distribution zones which supply this air directly to these penetrations. Neither the clarification in applicability of the LCO or the addition of three days to the out-of-service time allowed by these LCOs should significantly impact the availability of these systems to reduce containment leakage in the event of an accident. Since only a small portion of these systems are allowed to be temporarily out-of-service for a short period of time, there is little change in the probability that the WC&PPS and IVSWS will not be able, at least in part, to perform their function of reducing isolation valve or penetration leakage, if any should occur. In any event, the operability of these systems is not considered in previous evaluations. Therefore, no significant increase in the probability or consequences of an accident previously evaluated are involved.

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

Response

The proposed change does not involve a physical change to any plant systems, structures or components. The proposed change does not adversely affect the manner in which the plant is operated. 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

The LOCA offsite dose calculations, which do not assume WC&PPS and IVSWS operations, demonstrate that the calculated offsite doses are well within the 10 CFR Part 100 limits. Therefore, the margin of safety between the calculated offsite dose and the regulatory acceptable limits remains unchanged. However, operation of these systems assures that the containment leak rate is lower than that calculated by an uncalculated amount. This represents an additional assurance that the margin of safety remains unchanged. The revision of LCO applicability and out-of-service time for these systems will not significantly impact this additional assurance that containment leakage will be lower than that calculated. Since postulated LOCA assumptions remain unchanged and the proposed change does not involve a physical change to the WC&PPS and IVSWS, a significant reduction in the original margin of safety is not involved.

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, Director

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: December 20, 1988

Description of amendment request: The proposed amendments would delete from the Salem 1 and 2 Technical Specifications a portion of Surveillance Requirement 4.5.2.i associated with verifying that the Residual Heat Removal (RHR) System suction/isolation valves automatically close on a Reactor Coolant System pressure signal. Issuance of these amendments will allow the removal of the RHR Autoclosure Interlock (ACI) circuitry.

Basis for proposed no significant hazards consideration determination: Both the industry and the NRC have recognized the safety benefits of removing the Residual Heat Removal Autoclosure Interlock circuitry (RHR ACI). The NRC-AEOD case study on long term decay heat removal, Case Study Report AEOD/C503, Decay Heat Removal Problems at U.S. Pressurized Water Reactors, December 1985, recommended that consideration should be given to removal of the RHR ACI circuitry to minimize loss of decay heat removal events. Also, a study performed for the NRC by Brookhaven National Laboratory, NUREG/CR-5015, Improved Reliability of Residual Heat Removal Capability in PWRs as Related to Resolution of Generic Issue 99, May 1988, listed several improvements to reduce the risk of loss of decay heat removal. One improvement was the removal of the RHR ACI circuitry.

In parallel with the NRC activities, the Westinghouse Owners Group initiated a program to evaluate the removal of the RHR ACI circuitry on all Westinghouse designed plants. The end product of this program was WCAP-11736, Residual Heat Removal System Autoclosure Interlock Deletion Report for the Westinghouse Owners Group, Volumes 1 and 2, Revision 0.0, February 1988. WCAP-11736 documents the probabilistic analysis performed on the removal of the RHR ACI in terms of (1) the likelihood of an interfacing loss-of-coolant-accident (LOCA), (2) Residual Heat Removal system availability, and (3) low temperature over-pressurization concerns. The results of the analysis show that (1) the frequency of an interfacing system LOCA decreases with the removal of the RHR ACI, (2) removal of the RHR ACI increases the RHR system availability, and (3) removal of the RHR ACI has no effect on heat input transients; but will result in a small, but not significant, increase in the frequency of occurrence for some types of mass input transients with a decrease in others. The net effect of RHR ACI deletion is an improvement in safety.

To provide assurance that the Reactor Coolant system (RCS) will not be pressurized with the Residual Heat Removal system inlet valves open WCAP-11736 requires that a safety grade alarm be added that will actuate in the control room given a "VALVE NOT FULLY CLOSED" signal in conjunction with a "RCS PRESSURE-HIGH" signal. The intent of this alarm is to alert the operator that the RCS/RHR series suction/isolation valve(s) is(are) not fully closed, and that double valve isolation from the Reactor Coolant system to the Residual Heat Removal system is not being maintained. WCAP-11736 further states that applicable operating procedures should be modified to reflect this new alarm and describe the appropriate response. The licensee has committed to adding the alarm and modifying the operating procedures before implementing the requested technical specification change.

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:

The proposed change does not involve a significant hazards consideration because operation of Salem Generating Station Units 1 and 2 in accordance with this change would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The deletion of the RHR ACI was analyzed in WCAP-11736 in terms of (1) the frequency of an interfacing LOCA, (2) the availability of the RHR system, and (3) the effect on overpressure transients.

With the removal of the ACI and addition of a control room alarm, the probabilistic risk analysis predicts a decrease in the frequency of interfacing LOCAs from 8.35E-07 to 5.77E-07/year, a decrease of approximately 31%.

The availability of the RHR system was analyzed in three phases: initiation, short term cooling, and long term cooling. The probabilistic analysis indicated that deletion of the RHR ACI has no impact on the failure probability for RHR initiation. During short term cooling (72 hours after initiation), RHR ACI deletion decreased the RHR failure probability by 13%, from 1.60E-02 to 1.40E-02. The long term cooling RHR failure probability was calculated to decrease by 67% from 3.60E-02 to 1.20E-02.

Appendix D of WCAP-11736 presents the analysis used to determine the effect of removal of the ACI on overpressurization transients. The analysis categorizes the types of initiating events, determines their frequency of occurrence, and then identifies the consequences of these occurrences both with and without the ACI feature. The result is a list of overpressure consequence categories with associated failure probabilities (see Reference 4 [WCAP-11736], Appendix D, Tables D-9, -10 and -11). For the charging/safety injection event, consequence frequencies increased on the order of 1.0E-10 shutdown year. This is an insignificant increase as the overall consequence frequency of the charging/safety injection event is 1.25E-01. Likewise, for the letdown isolation with RHR system operable case, one frequency category was increased on the order of 1.0E-11. Again this is insignificant when compared with the total frequency of these events of 1.25E-01. For the letdown isolation with RHR system isolated event, the overall consequence frequency was reduced from 4.45E-01 to 2.22E-01. This occurs because many spurious closures of the RHR isolation valves cause the isolation of letdown.

Removing the RHR ACI reduces the frequency of this event by approximately 50%. It is concluded that the removal of the RHR ACI circuitry has an insignificant impact on the frequency of overpressurization events at Salem Station.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The effect of an overpressure transient at cold shutdown conditions will not be altered by removal of the RHR ACI function. With or without the ACI function, the RHR system could be subject to overpressure for which the RHR relief valves must be relied upon to limit pressure to within RHR design parameters. While it is true that the ACI initiates an

automatic closure of the RHR suction/isolation valves on high RCS pressure, overpressure protection of the RHR system is provided by the RHR system relief valves and not by the slow acting suction/isolation valves that isolate the RHR system from the RCS. This is reflected in the Salem UFSAR, which states:

Isolation of the RHR System is achieved with two remotely-operated series stop valves in the line from the RCS to the RHR pump suction and by two check valves in series in each line from the RHR pump discharge to the RCS, plus a remotely-operated stop valve in each discharge line. Overpressure in the RHR System is relieved through a relief valve to the pressurizer relief tank in the RCS. (Reference 7) [Salem UFSAR Section 5.5.7.2, page 5.5-28, Revision 7]

The purpose of the ACI feature is to ensure that there is a double barrier between the RHR system and RCS when the plant is at normal operating conditions, i.e., pressurized and not in the RHR cooling mode. Thus the ACI feature serves to preclude conditions that could lead to a LOCA outside of containment due to operator error. The safety function of the ACI is not to isolate the RHR system from the RCS when the RHR system is operating in the decay heat removal mode.

There are several methods to ensure that there is a double barrier between the RHR system and the RCS when the plant is at normal operating conditions. First, plant operating procedures instruct the operators to isolate the RHR system during plant heatup. Second, an alarm that will be installed as part of this change would annunciate in the control room given a "VALVE NOT FULLY CLOSED" signal in conjunction with a "RCS PRESSURE-HIGH" signal. This alarm would alert operators that either the RH1 or RH2 valve is not fully closed, and that double isolation has not been achieved. In conjunction with this, operators will be trained using revised alarm response procedures to ensure they act to restore double isolation or return to a safe shutdown condition. Third, the open permissive interlock, which is not being removed, will prevent the opening of the RH-1 and RH-2 whenever the RCS pressure is greater than the RHR system design pressure.

Since relief valves prevent overpressurization of the RHR system during shutdown conditions and several methods are in place to ensure that the RHR system is isolated from the RCS during normal plant conditions, removal of the ACI does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) involve a significant reduction in a margin of safety. The RHR ACI function is not a consideration in a margin of safety in the basis for any Technical Specification. However, since the probabilistic analysis of WCAP-11736 indicates that the availability of the RHR system is increased with the removal of the ACI, overall safety has been increased.

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

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: December 30, 1988

Description of amendment request: The proposed amendments to the Salem Units 1 and 2 Technical Specifications would permit the use of a new fuel design, Vantage 5 Hybrid, in both Salem Units. Additional changes are proposed to reduce the flow measurement uncertainty allowance because of recent plant modifications and to eliminate the rod bow penalty based on new analysis methods applied during the Vantage 5 Hybrid safety analysis. Specifically the Salem Units 1 and 2 Technical Specifications would be revised as follows:

1. Bases - Change the W-3 correlation to W-3 (R-Grid) and add the WRB-1 correlation and design Departure from Nucleate Boiling Ratio (DNBR) limits for Vantage 5H fuel (V5H).
2. Modify Specification 3.1.3.3 to incorporate a new rod drop time of less than or equal to 2.7 seconds.
3. Modify Unit 1 and Unit 2 Specification 3.2.3 to delete the Rod Bow Penalty as a function of burnup in the F-Delta-H equation and delete Figure 3.2-3.
4. Modify Unit 1 and Unit 2 Specification 3.2.5 Table 3.2-1 to define the Reactor Coolant System flow limit, including uncertainties, to be 357,200 GPM.

Basis for proposed no significant hazards consideration determination: Proposed revisions 1 and 2 are being requested to allow for the implementation of an improved fuel design, Westinghouse Vantage 5H fuel (V5H). Rod drop times are increased because of an increased dashpot effect caused by a reduction in guide tube diameter.

Proposed revisions 3 and 4 are being requested to incorporate new evaluation methods for the effects of fuel rod bow on departure from nucleate boiling (DNB). The new methods provide a basis to eliminate unnecessary power distribution penalties and to simplify the specification. Consistency between the Unit 1 and 2 Technical Specifications is also achieved.

Proposed revisions 5 and 6 are being requested to clearly define the DNB flow parameter limit plus uncertainties based upon the plants current configurations (previously licensed resistance temperature detector (RTD) flow uncertainty reductions) and to achieve consistency between the Unit 1 and 2 Technical 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. DNBR Bases Definition, Increased Rod Drop Time and Elimination of Rod Bow Penalty [Items 1-3]

The evaluation considered the effects of the proposed Technical Specification changes on the following areas:

a. Nuclear, Thermal-hydraulic and Mechanical Fuel Assembly Design
b. Non-Loca Accidents
c. Loca Accidents

The above areas have been evaluated including the concurrent effects of V5H features, thimble plug deletion, loose parts in the RCS and up to 3.5% steam generator tube plugging. In addition, transition core effects (mixed core of V5H and the 17X17 Standard product) have been addressed. The analyses required for the evaluations were performed by Westinghouse using approved methods and procedures (Attachment 4) [Public Service Electric and Gas Co. letter to NRC dated December 30, 1988, Plant Safety Evaluation for Salem Units 1 and 2 Fuel Upgrade, Dated November 1988]. LOCA evaluations were performed using the 1978 Westinghouse large break LOCA model which is our current evaluation model of reference. The results of the LOCA evaluations will be reevaluated against the Westinghouse updated model as part of the reanalysis required by the Salem Unit 2 Schedular Exemption from 10 CFR 50.46(a)(1)(i) (Ref. letter from J. C. Stone, Project Manager, Office of Nuclear Reactor Regulation to S. E. Miltenberger, Vice President and Chief Nuclear Officer, PSE&G, dated November 1, 1988). PSE&G has reviewed and concurs with the Westinghouse analyses.

Operation of the Salem Units in accordance with the proposed Technical Specification changes:

a. Will not involve a significant increase in the probability or consequences of an accident previously evaluated for the Salem Units. The evaluations of the Nuclear, Thermal-hydraulic, and Mechanical design effects support the conclusion that the requested changes are within the design criteria established in the Updated Final Safety Analysis Report. Consequently, no new mechanisms have been introduced to increase the probability of an accident occurring. The accident evaluations (LOCA and NON-LOCA) exhibit results which maintain the confidence level in the physical integrity of the fission product boundaries as defined in the Updated Safety Analysis Report. Therefore, the consequences of the accidents do not increase.

b. Will not create the possibility of a new or different kind of accident from any accident previously evaluated for the Salem Units. The evaluations performed establish that the Updated Final Safety Analysis Report design criteria and system responses during normal and accident conditions are bounding with respect to the requested changes. Therefore, the changes will not affect the function of any protection system nor introduce hardware which is different in design criteria requirements.

c. Will not involve a significant reduction in a margin of safety. The evaluations performed by Westinghouse addressed all design criteria and accident analyses. In performing the evaluations, the safety limits established by the Updated Final Safety Analysis Report and Technical Specifications were not modified such as to reduce the difference between the safety limit and the limit defined as the failure point of a fission product boundary. Therefore, the margins which were assumed in the accident analyses remain bounding for the proposed changes.

2. Definition of DNB Parameter Reactor Coolant Flow Limit [Item 4]

The evaluation considered the effect of the proposed Technical Specification changes on the following areas:

a. Updated Final Safety Analysis Report Chapter 15 Events

b. Protection System Setpoints and Response

The analyses required for the above evaluations were performed by Westinghouse and the results documented in WCAP-11579 (forwarded via PSE&G letter NLR-N87157, dated September 17, 1987). PSE&G has reviewed the WCAP and concurs with the results. In addition, a review of the units instrumentation uncertainties provides the conclusion that the results of WCAP-11579 are applicable for the Salem units. Specifically, the Unit 2 actual measurement uncertainties were verified to be bounded by the uncertainties assumed in WCAP-11579 (PSE&G letter NLR-N88171, dated October 19, 1988). The instrumentation in Unit 1 is comparable to the Unit 2 instrumentation, therefore, the comparison of uncertainties provided in NLR-N88171 is bounding for the Unit 1 instruments.

Operation of the Salem Units in accordance with the proposed Technical Specification changes:

a. Will not involve a significant increase in the probability or consequences of an accident previously evaluated for the Salem Units. The reduction in the uncertainty value is attributed to the reduced error associated with the modified RCS narrow range temperature monitoring system. The Chapter 15 accident analyses impacted by this modification were previously reviewed and approved by the NRC as Amendments 84 and 56 to the Salem Unit 1 and 2 licenses, respectively, and by Amendment 64 to the Unit 2 license.

b. Will not create the possibility of a new or different kind of accident from any accident previously evaluated for the Salem Units. The correction factor which modifies the RCS minimum flow value limit is based on an analysis of flow measurement uncertainties. The correction does not affect any process variable which inputs to a process control or reactor protection system control function. Therefore, Chapter 15 analyses are not affected.

c. Will not involve a significant reduction in a margin of safety. An RCS Flow uncertainty error of 3.5% was originally assumed for the purpose of calculating a minimum allowable RCS flow rate for safe plant operation. The uncertainty correction provides a reference point from which the relative magnitude of the safety margin between measured flow rate and design thermal flow rate can be inferred. WCAP-11579 demonstrates that the total uncertainty associated with the modified RCS narrow range temperature monitoring system could be reduced to a conservative value of 2.2% from existing value of 3.5%. In addition to the 2.2%, an additional uncertainty of 0.1% for feedwater venturi fouling will be added for a total uncertainty factor of 2.3%. The evaluations provided show that the change to the allowable flow uncertainty does not result in a reduction to the margin of safety as identified in the Final Safety Analysis Report. The value of the thermal design flow used in DNBR analyses remains the same as in the current UFSAR.

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

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: January 3, 1989

Description of amendment request: The proposed amendments would define for Salem Unit 1 and Salem Unit 2 the Fully Withdrawn position of Rod Cluster Control Assemblies to address potential rod wear concerns as seen at other Westinghouse designed plants. Sections of the Salem Unit 1 and Salem Unit 2 Technical Specifications that are affected by the definition of Fully Withdrawn are to be changed accordingly. In addition, changes are proposed to delete from Salem Unit 1 Technical Specifications a rod bank insertion limit curve for three loop operation and to correct inconsistencies between Salem Unit 1 and Salem Unit 2 Technical Specifications. Specifically the Same Unit 1 and Salem Unit 2 Technical Specifications would be revised as follows:

1. Definitions - add a definition for the fully withdrawn position of the Rod Cluster Control Assemblies (RCCAs).
2. Modify definition 1.28, Shutdown Margin, Specifications 3.1.3.4 and 3.10.1, and Bases 2.1.1 and 3/4.1.3 to incorporate the new definition of "Fully Withdrawn".
3. Replace Figure 3.1-1 to incorporate the new definition of Fully Withdrawn.
4. Delete Figure 3.1.2 from Unit 1.
5. Modify Specification 3.1.3.3 to clarify rod drop test requirements.
6. Modify Unit 2 Specification 3.1.3.2.2 to incorporate the rod drop testing requirements previously in Specification 3.10.5.
7. Add to Unit 1 Specification 3.1.3.2.2 rod drop test requirements as included in Unit 2 to achieve consistency between units.
8. Delete Unit 2 Specification 3.10.5.

Basis for proposed no significant hazards consideration determination: Proposed revision items one through three are being requested to address potential rod wear concerns as seen previously at other Westinghouse plants. These items redefine Fully Withdrawn to be between 222 and 228 steps withdrawn.

Proposed revision Item four is being requested to delete the curve implementing three loop operations which is not currently allowed but is still affected by redefining Fully Withdrawn. Rather than modifying this specification, it is proposed to be deleted. This is consistent with the Unit 2 specifications.

Item five is being requested to clarify that rod drop test times are to be performed from 228 steps withdrawn. With the proposed redefinition of Fully Withdrawn, test times could be performed from 222 steps withdrawn if this clarification was not made.

Proposed revisions six through eight are being requested to correct an inconsistency present in the current Unit 2 Technical Specifications. Previously, a change was approved that no longer required that the Analog Rod Position Indication (ARPI) be operable in Modes 3, 4 and 5. This eliminates the need for Specification 3.10.5 since the other requirements are being addressed in specification 3.1.3.2.2. The rod drop test requirements are being added to Unit 1 for consistency between units.

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. Rods Fully Withdrawn Definition (Items 1-5)

A safety evaluation has been performed to address repositioning the fully withdrawn position of the RCCAs (Attachment 4) [Public Service Electric and Gas Co. Letter to NRC dated January 3, 1989, Analysis of effects of RCCA Repositioning on loss-of-coolant related accidents]. The evaluation considered the effects of the proposed technical specification changes on the following areas:

a. Small Break LOCA
b. Large Break LOCA
c. Short and Long Term LOCA
d. Steam Generator Tube Rupture
e. Post-LOCA Long Term Cooling
f. Hot Leg Switchover to Prevent Potential Boron Precipitation
g. Blowdown Reactor Vessel and Loop Forces
h. Non-LOCA Transients

The conclusions of the evaluation are as follows:

a. The changes in the definition of the fully withdrawn RCCA position proposed create no significant changes in the affected safety parameters involved in verification of current technical specification limits. The involved safety parameters include those parameters normally addressed by the cycle specific Reload Safety Evaluation Checklist. The change of the fully withdrawn position from 228 steps to 222 steps or higher involves only a small amount of absorber being inserted into the active region of core and does not result in any design or regulatory limit being exceeded.

b. No FSAR safety limits are exceeded based on the proposed technical specification change. The position of the control and shutdown banks, relative to each other in the core will not change; therefore the limiting axial power distribution assumed for the DNB analyses remain applicable. The FSAR conclusion that the DNBR design basis acceptance criteria is met for the Condition II events remains valid. Additionally, there is no significant impact on any core physics assumptions and design peaking factors important to the non-LOCA safety analyses and the reload verification.

c. The proposed change does not invalidate current control rod drop times or other tripped rod characteristics assumed in the LOCA licensing basis analysis.

Operation of the Salem Units in accordance with this proposed technical specification change:

a. Would not involve a significant increase in the probability or consequences of an accident previously evaluated for the Salem Units, since the changes caused by repositioning the fully withdrawn position of the control rods are bounded by those assumed in the accident analyses.

b. Would not create the possibility of a new or different kind of accident from any accident previously evaluated for the Salem Units, since no plant hardware changes are required by this change.

c. Would not involve a significant reduction in a margin of safety, since the margin which was assumed in the accident analyses bounds the change proposed.

2. Elimination of Special Test Exemption 3.10.5 (Items 6-8)

Operation of the Salem Units in accordance with this proposed Technical Specification change:

a. Would not create a significant increase in the probability or consequences of an accident previously evaluated for the Salem Units since the change is administrative in that it eliminates an unnecessary specification and incorporates the requirements into an existing specification. Additionally, it imposes a like requirement into the Unit 1 Technical Specification;

b. Would not create the possibility of a new or different kind of accident from any accident previously evaluated for Salem since no plant hardware modifications are required and no tests are being deleted;

c. Would not involve a significant reduction in a margin of safety, since no analytical or test changes are being made.

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

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

Date of amendment request: December 30, 1988

Description of amendment request: The proposed amendment involves proposed changes to the Surveillance Standards of the Technical Specifications on a one-time basis. The requested changes are for a one-time extension for surveillances that are currently required by the Technical Specifications to be performed beginning March 29, 1989. Specifically, the changes involve the following Technical Specification sections:

4.0 General Surveillance Requirements

4.4.1.2 Local Leakage Rate Tests

4.5.3 Decay Heat Removal System and Reactor Building Spray System Leakage

The licensee requested that the surveillances be performed at the next refueling outage currently scheduled to begin on or before August 1, 1989.

This request encompasses all Hot Shutdown and Cold Shutdown surveillances due prior to August 1, 1989 except those regarding the emergency diesel generators. In addition this proposed amendment clarifies the surveillance period of the Decay Heat Removal Test defined in Specification 4.5.3.2.A.

All requested surveillance test extensions are associated with surveillances normally performed during refueling outages. Since the restart of Rancho Seco in March 1988, following an extended maintenance outage, the duration of the current refueling cycle, Cycle 7, has been lengthened due to operational testing at reduced power and several short maintenance outages.

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 determined that the requested amendment per 10 CFR 50.92 does not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated because extending surveillances by four months does not significantly affect the probability of accidents, nor will degradation occur in these four months that would change the consequences of an accident; or

(2) Create the possibility of a new or different kind of accident from an accident previously evaluated because the proposed Technical Specification changes do not change the operation of any equipment and the systems' abilities to perform their intended functions will not be altered; or

(3) Involve a significant reduction in a margin of safety because system operation is not affected and deferral of the surveillances will not result in significant degradation of equipment.

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

Accordingly, the Commission has proposed to determine that the above changes do not involve a significant hazards consideration.

Local Public Document Room location: Martin Luther King Regional Library, 7340 24th Street Bypass, Sacramento, California 95822

Attorney for licensee: David S. Kaplan, Sacramento Municipal Utility District, 6201 S Street, Post Office Box 15830, Sacramento, California 95813

NRC Project Director: George W. Knighton

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: December 29, 1988

Description of amendment request: The proposed amendment is a request to revise Appendix A Technical Specifications to incorporate Limiting Conditions for Operation (LCOs) and Surveillance requirements associated with the containment spray actuation instrumentation. In accordance with resolution to Systematic Evaluation Program Topic VI-10.A, "Testing of Reactor Trip System and Engineered Safety Features, Including Response Time Testing," this proposed change incorporates LCOs and surveillances that are not currently included in the technical specifications.

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

The Containment Spray Actuation System (CSAS) is an accident mitigation system with no impact on accident probabilities. The CSAS is an existing system and this proposed change will incorporate surveillance and operability requirements into the technical specifications. The operability of the CSAS does affect previously analyzed accident consequences, as these accidents require successful operation of the CSAS to achieve their calculated design basis conclusion. Therefore, it is concluded that operation of the facility in accordance with this proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The CSAS is an existing plant system and formally requiring its operability and surveillance does not create any new or different accidents. The proposed LCOs and surveillance requirements are consistent with STS specifications in this area, and, accordingly, are appropriate. Therefore, it is concluded that operation of the facility in accordance with this proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

RESPONSE: No

Requiring the CSAS to be operable and surveilled will preserve existing, analyzed margins of safety. As the proposed change is in conformance with STS guidance, a required and assumed margin of safety will be maintained. Therefore, it is concluded that operation of the facility in accordance with this proposed change does not involve a significant reduction in a 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

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: December 29, 1988

Description of amendment request: The proposed amendment would revise Technical Specifications associated with the Reactor Protection System instrumentation. This proposed change incorporates Limiting Conditions for Operation and Surveillance requirements into the technical specifications that are currently performed by procedure. In addition, surveillance intervals and out of service times have been increased in accordance with Westinghouse recommendations as documented in WCAP-10271.

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

Implementation of the proposed changes is expected to result in an acceptable increase in total Reactor Protection System yearly unavailability. This increase, which is primarily due to less frequent surveillance testing, results in an increase of similar magnitude in the probability of an Anticipated Transient Without Scram (ATWS) and in the probability of core melt resulting from an ATWS. Based on the following, these slight increases are judged to be acceptable.

Implementation of the proposed changes is expected to result in a significant reduction in the probability of core melt from inadvertent reactor trips. This is a result of a reduction in the number of inadvertent reactor trips (0.5 fewer inadvertent reactor trips per unit per year) occurring during testing of RPS instrumentation. This is primarily attributable to testing in bypass and less frequent surveillance.

The reduction in inadvertent core melt probability is sufficiently large to counter the increase in ATWS core melt probability resulting in an overall reduction in total core melt probability. Incorporation of additional controls not currently in the technical specifications does not impact the probability or consequences of an accident previously evaluated, as these additional surveillances are currently maintained administratively by plant procedures.

The proposed changes do not result in an increase in the severity or consequences of an accident previously evaluated. Implementation of the proposed changes affects the probability of failure of the RPS but does not alter the manner in which protection is afforded nor the manner in which limiting criteria are established.

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

The proposed changes do not result in a change in the manner in which the Reactor Protection System provides plant protection. No change is being made which alters the functioning of the Reactor Protection System (other than in a test mode). Rather, the likelihood or probability of the Reactor Protection System functioning properly is affected as described above. Therefore, the proposed changes do not create the possibility of a new or different kind of accident.

The proposed changes do not involve hardware changes except those necessary to implement testing in bypass. Some existing technical specifications allow testing in bypass. Testing in bypass is also recognized by IEEE Standards. Therefore, testing in bypass has been previously approved and implementation of the proposed changes for testing in bypass does not create the possibility of a new or different kind of accident from any previously evaluated. Furthermore since the other proposed changes do not alter the functioning of the RPS, the possibility of a new or different kind of accident form any previously evaluated has not been created.

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

Response: No

The proposed changes do not alter the manner in which safety limits, limiting safety system setpoints or limiting conditions for operation are determined. The impact of the reduced testing other than as addressed above is to allow a longer time interval over which instrument uncertainties (e.g., drift) may act. Experience at two Westinghouse plants with extended surveillance intervals has shown the initial uncertainty assumptions to be valid for reduced testing.

Implementation of the proposed changes is expected to result in an overall improvement in safety by:

a. 0.5 fewer inadvertent reactor trips per unit. This is due to less frequent testing and testing in bypass which minimizes the time spent in a partial trip condition.

b. Higher quality repairs leading to improved equipment reliability due to longer repair times.

c. Improvements in the effectiveness of the operation staff in monitoring and controlling plant operation. This is due to less frequent distraction of the operator and shift supervisor to attend to instrumentation testing.

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

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

Date of amendment requests: September 29, 1988 (TS 257)

Description of amendment requests: Browns Ferry Nuclear Plant Technical Specifications Tables 3.2.J and 4.2.J, Seismic Monitoring Instrumentation, are being revised to reflect the manufacturer's suggested testing for the upgraded triaxial peak accelerographs. This upgrade replaced the Terra Technology (PRA-103S) seismic instruments with the EngDahl (PAR-400-2) seismic instruments. These new instruments were installed to improve instrument efficiency and dependability. In addition to the manufacturer's recommendations, several administrative changes are also being made to these tables and to the Bases for Technical Specification, Section 3.2.

Specifically, the channel calibration frequency for triaxial time history accelerographs and the triaxial peak accelerographs would be changed from "N/A" to "R" (refueling). The channel functional test frequency for the triaxial peal accelerographs would be changed from "12 months" to "N/A." The channel functional test frequency for the triaxial time history accelerographs and the biaxial seismic switches would be changed from "six months" to "SA" (semi-annually). The channel calibration frequency for biaxial seismic switches would be changed from once/operating cycle to "R"; i.e., each refueling cycle. The note which says "except seismic switches" and is referenced by the channel check requirements for the triaxial time history accelerographs and the biaxial seismic switches would be deleted. The other administrative changes would provide a consistent order to the tables, numbering the table entries for each type of instrument, and correcting the spelling of accelerograph. Also, in each table after each biaxial siesmic switch, the correct elevation (EL. 519) is added.

Basis for proposed no significant hazards consideration determination: The Commission has provided Standards for determining whether a significant hazards determination exists as stated in 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 previously evalauted or (3) Involve a significant reduction in a margin of safety. The staff has reviewed the licensee's no significant hazards determination analyses, provided to the Commission, in accordance with 10 CFR 50.91. The staff concurs with the licensee's determination. However, the staff has determined that additional clarification was needed and, therefore, the staff is providing the following determination with these clarifications:

1. The replacement of the original seismic instruments with the EngDahl instruments does not involve a significant increase in the probability or consequences of an accident previously evaluated. Regulatory Guide 1.12 requires that seismic instrumentation be installed at nuclear power plants so that, in the event of an earthquake, the seismic response of plant features important to safety can be determined promptly. This response is then compared with that used in the design basis in order to decide whether the plant can continue to be operated safety. Although the monitoring instrumentation hardware is being changed, the intended monitoring functions and data provided by the EngDahl instruments are consistent with the appropriate Regulatory Guide. The replacement of the seismic instruments will provide easier field calibrations to be performed, greater reliability than the previous instruments, and therefore improve plant ability to monitor peak accelerations during a seismic event. The replacement of these instruments support the current design bases, noted regulatory requirements, and does not invalidate any safety analysis assumed for the licensing and operation of BFN.

The surveillance requirements in Table 4.2.J are being revised to incorporate the vendor recommended testing frequencies. The revision to the Channel Calibration testing frequencies for the triaxial history accelerographs and triaxial peak accelerographs to once per refueling outage is consistent with the GE Standard Technical Specifications as well as Table 1, Frequency of Maintenance, of ANSI/ANS-2.2-1978, "Earthquake Instrumentation Criteria for Nuclear Power Plants." The addition of these surveillances provides added assurance that the subject equipment performs as designed. Deleting the triaxial peak accelerograph Channel Functional Test and adding the Channel Calibration Test does not degrade the intent of the current TS since the Channel Calibration test is a more comprehensive operability verification. The changes made to the surveillance testing frequencies will still provide adequate verification that the instrumentation is performing its intended design function.

The administrative changes being made are to correct typographical errors existing in the current Tables. The other administrative changes provide greater consistency between the two Tables, make the testing frequency notations consistent with the existing definitions section, and provide elevations for the location of the seismic monitors.

The changes discussed above do not affect the function or intended design bases for any safety-related equipment currently installed at BFN. The replacement instrumentation, amended surveillance testing, nor the administrative changes do not change any of the safety analysis, assumptions made in the Final Safety Analysis Report, or calculations used in the design or licensing basis for BFN.

2. The proposed change does not create the possibility of a new or different kind of accident from an accident previously evaluated. The replacement EngDahl seismic instruments provide the same type of data and are similar in size to the original instruments. The seismic instruments are mounted on specific pipes inside the plant.

The size of the replacement instruments are similar enough that only minor mounting bracket modifications were needed. The seismic qualification of the piping was not affected. Since this is a hardware modification, the intended function and parameters monitored will remain the same as the original instruments. This amendment does not change the intended function or operation of any safety-related equipment, emergency operating procedures, or operating procedures, or operating practices.

Amending the surveillance frequencies as noted is in compliance with the appropriate industry standards and vendor recommendations. This amendment does not change the intent of the existing TS and additionally ensures that, through the proper testing and calibration, the instrumentation is performing its intended function.

The proposed administrative changes provide consistency between the Tables. These changes do not affect any operational conditions, safety-related equipment, or setpoints which could cause or adversely affect the mitigation of a new or different kind of accident from an accident previously evaluated.

3. The proposed changes do not significantly decrease the margin of safety at BFN. The replacement of the seismic instruments is a hardware change only. The new instruments will provide added reliability and therefore, improve the plant's overall ability to monitor peak accelerations caused by a seismic event. The replacement instruments will perform the same function as the original seismic instruments.

Amending the surveillance frequencies as noted is consistent with current industry standards and practices. These changes are also consistent with the vendor recommendations. The surveillances are to be utilized to ensure appropriate instrument function.

The administrative changes are being made to provide consistency between the Tables and correct typographical errors. These changes are administrative in nature and do not reduce any margin of safety.

The seismic monitoring instrumentation is not required to mitigate the consequences of any design basis events, but rather provide data for evaluation after a seismic event to ensure that the plant can continue to operate safely. Therefore, the proposed TS does not involve a significant reduction in the margin of safety.

Therefore, the staff proposed to determine that the application for amendment involve no significant hazards consideration.

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

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

NRC Assistant Director: Suzanne Black

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

Date of amendment requests: December 22, 1988 (TS 88-34)

Description of amendment requests: The Tennessee Valley Authority (TVA) proposes to modify the Sequoyah Nuclear Plant (SQN) Units 1 and 2 Technical Specifications (TS). The changes are to remove inappropriate testing requirements associated with the auxiliary building gas treatment system (ABGTS). Surveillance requirements for ABGTS activation exist in Section 7, "Plant Systems," and Section 9, "Refueling Operations," of the TS. These requirements are TS 4.7.8.d.2 and 4.9.12.d.2. The ABGTS surveillance requirements from Section 7 are applicable during Modes 1, 2, 3, and 4; and the ABGTS surveillance requirements from Section 9 are applicable whenever irradiated fuel is in the spent fuel storage pool. The ABGTS test requirement associated with the auxiliary building ventilation monitoring systems (ABVMS) would be deleted from both Sections 7 and 9. The ABGTS test requirement associated with a phase A containment isolation signal would be deleted in Section 9 but would remain in Section 7. The ABGTS test requirement associated with the high radiation signal from the spent fuel pool monitors would be deleted in Section 7 but would remain in Section 9.

A new requirement has been added to Table 4.3.9 of Specification 3.3.3.10, "Radioactive Gaseous Effluent Monitoring," to demonstrate automatic isolation of the auxiliary building ventilation exhaust any time the ABVMS (radiation monitor) indicates measured levels above the alarm/trip setpoint. This requirement is currently in Sections 7 and 9 as part of the ABGTS actuation test for a high radiation signal from the ABVMS but would be deleted from Sections 7 and 9. Also, two typographical errors in the Unit 1 Specification 3.3.3.10 have been corrected.

Basis for proposed no significant hazards consideration determination: TVA provided the following information on the ABGTS which is part of the auxiliary building ventilation system (ABVS) in its submittal on the proposed TS changes.

The ABVS is described in section 9.4.2 of the Final Safety Analysis Report (FSAR. This system serves all areas of the auxiliary building including the radwaste areas and the fuel handling areas. It is designed to maintain acceptable environmental conditions for personnel access, for protection of mechanical and electrical equipment and controls, and to limit the release of radioactivity to the environment.

The current ABGTS surveillance requirements impose appropriate actions under certain conditions. For example, should the single auxiliary building vent radiation monitor become inoperable, ABGTS must be declared inoperable and consequently a plant shutdown is required by Specification 3.0.3. Similar effluent monitoring technical specifications allow continued reactor operation with vent path sampling. Similar inappropriate action applies to inoperability of the fuel pool monitors while in modes 1, 2, 3, and 4. An inoperable fuel pool radiation monitor, while in these modes, would require that ABGTS be declared inoperable and could possibly result in a plant shutdown. The more appropriate action is to limit crane operation with loads over the spent fuel pit as specified in Technical Specification 3.9.12.

Another inappropriate action would exist in Mode 6 with the Phase A containment isolation signal becoming inoperable. Crane operation with loads over the spent fuel pit may be prohibited when, in fact, the loss of coolant accident (LOCA) mitigation equipment is not required. The proposed technical specification change will alleviate these problems by assigning each ABGTS surveillance requirement to its proper accident signal.

Deletion of the ABGTS actuation surveillance requirement from the high radiation signal in the auxiliary building vent will significantly reduce the amount of surveillance work, system alignment, and unnecessary operator interface required to perform the test. The current test addresses all aspects of the ABGTS function: ABGTS filter train start, auxiliary building isolation, ABSCE (auxiliary building secondary containment enclosure) establishment, and accident mode room cooling.

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 technical specification 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 ABGTS is an engineered safety features system required to function postaccident. The signal for ABGTS initiation on high radiation in the auxiliary building vent is not included in any accidents evaluated by the safety analysis report. Deletion of the subject test requirement has no impact on the function of the ABGTS or the radiation monitor itself. Deletion of [the surveillance requirements associated with] the phase A containment isolation signal and the fuel handling area radiation monitor signal is consistent with assumptions made in the accident analysis. The typographical corrections are strictly administrative and do not alter any intent of the specification. Therefore, there is no change 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. High radiation in the auxiliary building vent initiates an ABGTS start and isolation of the auxiliary building. For any accident where ABGTS is assumed, the start signal would be provided by redundant channels in the initiation logic, all of which are safety-grade, trained redundant instruments. The phase A signal and the fuel handling area signal are required operable as assumed in the FSAR. The typographical corrections are strictly administrative. Thus, the possibility of a new or different kind of accident has not been created.

(3) Involve a significant reduction in a margin of safety. No change is being made to the hardware or function of ABGTS or the auxiliary building vent monitor. The actual testing of the phase A signal and the fuel handling area signal is not changed. Because of the test signal being deleted is backed up by redundant channels, which are safety-grade, trained, and therefore more reliable, no margin of safety is reduced. The typographical corrections are strictly administrative.

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

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

Date of amendment request: November 29, 1988, supplemented November 30, 1988.

Description of amendment request: The amendment would reflect personnel changes, correct typographical errors, and make minor word changes to clarify the intent of Technical Specifications (TS).

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (51 FR 7751) of actions that are considered not likely to involve significant hazards considerations. Example (i) of this guidance states: "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 directly related to the example. They do not involve a decrease in management support or involvement in the Kewaunee Plant. Engineering and technical support supplied by the plant and corporate staff would not be decreased as a result of the changes. The proposed changes are purely administrative and editorial.

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

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

Date of amendment request: August 9, 1988

Brief description of amendment: The proposed license amendment would allow a one-time extension of the surveillance intervals for certain surveillances normally performed with the unit shutdown. The extensions involve:

1. ice basket weighing;
2. ice condenser flow passage inspections;
3. ice condenser inlet door testing; and
4. resistance temperature detector calibrations.

Date of publication of individual notice in **Federal Register**: January 17, 1989 (54 FR 1806)

Expiration date of individual notice: February 16, 1989.

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

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

Date of amendment request: November 15, 1988

Brief Description of amendment request: The proposed amendment would change the Technical Specifications to reflect a revised safety analysis that includes the use of fuel designed and fabricated by Advanced Nuclear Fuels Corporation.

Date of publication of individual notice in **Federal Register**: January 24, 1989 (54 FR 3545)

Expiration date of individual notice: February 23, 1989.

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

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.

Arizona Public Service Company, et al., Docket No. STN 50-528 Palo Verde Nuclear Generating Station, Unit 1 Maricopa County, Arizona

Date of application for amendment: July 25, 1988

Brief description of amendment: The Amendment revises TS Section 3.3.2, Table 3.3-5, "Engineered Safety Features Response Times" by clarifying the response time requirements for radiation detectors associated with Control Room Essential Filtration Actuation. Minor editorial corrections have also been incorporated in TS Section 3/4.3.2, "Engineered Safety Features Actuation System Instrumentation."

Date of issuance: December 28, 1988

Effective date: December 28, 1988

Amendment No.: 41

Facility Operating License No. NPF-41: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 21, 1988 (53 FR 36666). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 28, 1988.

No significant hazards consideration comments received: No.

Local Public Document Room location: Phoenix Public Library, Business and Science Division, 12 East McDowell Road, Phoenix, Arizona 85004.

Arizona Public Service Company, et al, Docket No. STN 50-528, Palo Verde Nuclear Generating Station, Unit 1, Maricopa County, Arizona

Date of application for amendment: December 23, 1988

Brief description of amendment: The amendment revised Surveillance Requirement 4.1.3.1.2 to allow continued operation of PVNGS Unit 1, until the end of the current cycle (approximately 3 months), without conducting any further exercise tests of control element assembly (CEA) No. 64.

Date of issuance: January 13, 1989

Effective date: January 13, 1989

Amendment No.: 42

Facility Operating License No. NPF-41: Amendment changed the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: Yes (54 FR 75 dated January 3, 1989). That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by January 18, 1989, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 13, 1989, which makes a final no significant hazards consideration determination.

Attorney for Licensee: Arthur C. Gehr, Esq., Snell & Wilmer, 3100 Valley Center, Phoenix, Arizona 85007.

Local Public Document Room location: Phoenix Public Library, Business and Science Division, 12 East McDowell Road, Phoenix, Arizona 85004.

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

Date of application for amendments: March 6, 1987 supplemented January 6 and March 9, 1988 and January 6, 1989.

Description of amendments: These amendments revise the LaSalle County Station, Units 1 and 2 Technical Specifications by removing all references to the ammonia detector monitoring instrument system.

Date of issuance: January 18, 1989

Effective date: January 18, 1989

Amendment Nos.: 61 and 42

Facility Operating License Nos. NPF-11 and NPF-18. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 8, 1987 (52 FR 11357). The supplemental submittals by the licensee provided further revisions to the initial probability analysis, but did not change the staff's initial determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 18, 1989.

No significant hazards consideration comments received: No

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

Commonwealth Edison Company, Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: September 16, and November 18, 1988

Brief description of amendments: Revise Main Steam Line Radiation Monitors trip setpoint for reactor protection system from seven times normal full power background to 15 times. This is necessary to provide for implementation of Hydrogen Water Chemistry control.

Date of issuance: January 18, 1989

Effective date: January 18, 1989

Amendment Nos.: 112 and 108

Facility Operating License Nos. DPR-29 and DPR-30. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 14, 1988 (53 FR 50321). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 18, 1989.

No significant hazards consideration comments received: No

Local Public Document Room location: Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

Commonwealth Edison Company, Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: November 16, 1988

Brief description of amendments: Revises surveillance interval of Main Steam Isolation Valves local leak rate testing from 18 months to each fuel cycle, not to exceed once every 24 months.

Date of issuance: January 19, 1989

Effective date: January 19, 1989

Amendment Nos.: 113 and 109

Facility Operating License Nos. DPR-29 and DPR-30. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 14, 1988 (53 FR 50322). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 19, 1989.

No significant hazards consideration comments received: No

Local Public Document Room location: Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

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

Date of application for amendment: November 7, 1988

Brief description of amendment: The amendment revises Table 7.2-1, "Radioactive Liquid Effluent Monitoring Instrumentation" and Table 8.2-1, "Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirement" by providing the respective radiation monitors identification label with the previous identified radiological monitoring locations.

Date of Issuance: January 24, 1989

Effective date: January 24, 1989

Amendment No.: 111

Facility Operating License No. DPR-61. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated January 24, 1989.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 24, 1988.

Brief description of amendment: This amendment revises the Fermi-2 Technical Specifications to remove the organization charts from the Technical Specifications following the guidance provided in NRC Generic Letter 88-06. The amendment also makes various administrative changes to Section 6.0 of the Technical Specifications.

Date of issuance: January 24, 1989

Effective date: January 24, 1989

Amendment No.: 30

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

Date of initial notice in **Federal Register:** August 10, 1988 (53 FR 30129). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 24, 1989.

No significant hazards consideration comments received: No.

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

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

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

Date of application for amendments: September 3, 1987, as supplemented February 27, September 9, and September 20, 1988.

Brief description of amendments: The amendments revised the Technical Specifications to replace the values of cycle-specific parameter limits with a reference to the Core Operating Limits Report which contains the values of those limits.

Date of issuance: January 26, 1989

Effective date: January 26, 1989

Amendment Nos.: 172, 172, and 169

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

Date of initial notice in **Federal Register:** December 14, 1988 (53 FR 50325). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 26, 1989.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: November 12, 1986, supplemented by letter dated November 17, 1988

Brief description of amendment: The amendment revises the visual inspection requirements for snubbers and the service life monitoring requirements.

Date of issuance: January 23, 1989

Effective date: January 23, 1989

Amendment No. 135

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

Date of initial notice in **Federal Register:** March 25, 1987 (52 FR 9567). The November 17, 1988 submittal provided additional clarifying information and did not change the determination of the initial notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 23, 1989

No significant hazards consideration comments received: No

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

Duquesne Light Company, Docket No. 50-412, Beaver Valley Power Station, Unit No. 2, Shippingport, Pennsylvania

Date of application for amendment: August 30, 1988, supplemented by letter dated November 10, 1988

Brief description of amendment: The amendment changes the Technical

Specifications to allow storage of fuel and spent fuel assemblies up to enrichment of 4.85 weight-percent U-235.

Date of issuance: January 17, 1989

Effective date: January 17, 1989

Amendment No. 12

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

Date of initial notice in **Federal Register**: October 5, 1988 (53 FR 39168). The November 10, 1988 submittal provided additional clarifying information and did not change our initial determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 17, 1989

No significant hazards consideration comments received: No

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

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50-424, Vogtle Electric Generating Plant, Unit 1, Burke County, Georgia

Date of application for amendment: December 6, 1988

Brief description of amendment: The amendment modified the Technical Specifications to raise the minimum diesel generator voltage for tests not requiring circuit breaker closure to ensure that the generator "ready-to-load" condition is met during surveillance.

Date of issuance: January 23, 1989

Effective date: January 23, 1989

Amendment No.: 16

Facility Operating License No. NPF-68: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 15, 1988 (53 FR 50480). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 23, 1989.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: November 30, 1988, as supplemented by letter dated December 12, 1988

Brief description of amendment: The amendment deletes the requirement in Technical Specification, Table 3.1.1.A.8 for a Low Condenser Vacuum Scram when the Reactor Mode Switch is in the refuel position. This change clarifies the Technical Specification to allow Rod Scram time testing to be performed while shutdown. The amendment also revises Technical Specification, Table 3.1.1.C.1 to add a reference to note "11" in the startup mode for the High Reactor Pressure Isolation Condenser initiative function. This change is necessary to install new analog pressure sensors during refueling outage 12R.

Date of Issuance: January 13, 1989

Effective date: January 13, 1989

Amendment No.: 131

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

Date of initial notice in **Federal Register**: December 12, 1988 (53 FR 49943). The December 12, 1988 submittal corrected a Technical Specification page and did not change the determination of the initial notice. The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated January 13, 1989

No significant hazards consideration comments received: No.

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

Maine Yankee Atomic Power Company, Docket No. 50-309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of application for amendment: March 5, 1987 as clarified by letters dated October 11, 1988 and November 1, 1988.

Brief description of amendment: This amendment to the license updates the physical security plan.

Date of issuance: January 23, 1989

Effective date: January 23, 1989

Amendment No.: 110

Facility Operating License No. DPR-36. Amendment revised a license condition.

Date of initial notice in **Federal Register**: (53 FR 50331) December 14, 1988. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 23, 1989

No significant hazards consideration comments received: No.

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, Maine 04578.

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

Date of application for amendment: January 14, 1988

Brief description of amendment: To eliminate a contradiction between Technical Specification 3.1.1.b(3)(b) and Specification 3.1.1.e and to require verification in Specification 3.1.1.b(3)(b) that the control rod program is being followed appropriately.

Date of issuance: January 26, 1989

Effective date: January 26, 1989

Amendment No.: 103

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

Date of initial notice in **Federal Register**: December 14, 1988 (53 FR 50332). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 26, 1989

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 11, 1988

Brief description of amendment: The amendment changes Technical Specification (TS) 4.6.1.2, "Containment Leakage," to allow the use of the "mass point" methodology, per ANSI/ANS 56.8-1981 and 10 CFR Part 50, Appendix J, Section III. A(3), in addition, or as an alternative to, the "total time" methodology currently specified in the TS.

Date of issuance: January 17, 1989

Effective date: January 17, 1989

Amendment No.: 30

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

Date of initial notice in **Federal Register**: September 21, 1988 (53 FR 3662). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 17, 1989

No significant hazards consideration comments received: No.

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

Philadelphia Electric Company, Docket No. 50-352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of application for amendment: November 1, 1988

Brief description of amendment: The amendment changed the Technical Specifications to reflect NRC approved modifications to certain containment penetrations to permit foward leak testing of associated isolation valves and testing of valve packing leakage.

Date of issuance: January 18, 1989

Effective date: 60 days after date of issuance

Amendment No. 15

Facility Operating License No. NPF-39. This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 14, 1988 (53 FR 50334). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 18, 1989.

No significant hazards consideration comments received: No

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

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: January 19, 1988

Brief description of amendments: These amendments changed the Technical Specifications and surveillance requirements applicable to containment hydrogen analyzers.

Date of issuance: January 25, 1989

Effective date: As of the date of issuance with implementation to be completed within 30 days of the date of issuance, for both units.

Amendment Nos. 90 and 65

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

Date of initial notice in **Federal Register:** August 24, 1988 (53 FR 32295). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 25, 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 No. 50-206, San Onofre Nuclear Generating Station, Unit No. 1, San Diego County, California

Date of application for amendment: March 20, 1987, as supplemented July 22, 1988.

Brief description of amendment: The amendment allows a seal leakage test to be performed in lieu of a full pressure test on the containment air lock when no maintenance has been performed on the air lock that could affect sealing capability of the air lock. The amendment also makes two editorial clarifications to the testing requirements on air lock doors.

Date of issuance: January 24, 1989

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

Amendment No.: 118

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

Date of initial notice in **Federal Register:** September 7, 1988 (53 FR 34611). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 24, 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.

NRC Project Director: George W. Knighton

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

Date of application for amendments: September 29, 1988 (TS 255)

Brief description of amendments: The amendments modify Technical Specifications Sections 3.6.H and 4.6.H to permit removal of references to seismic restraints and supports.

Date of issuance: January 19, 1989

Effective date: January 19, 1989, and shall be implemented within 60 days

Amendments Nos.: 163, 160, and 134

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

Date of initial notice in **Federal Register:** October 19, 1988 (53 FR 41001). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 19, 1989.

No significant hazards consideration comments received: No

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

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

Date of application for amendments: August 4, 1988 (TS 252)

Brief description of amendments: These amendments add Technical Specifications Limiting Conditions for Operation and Surveillance Requirements for the Anticipated Transients Without Scram (ATWS) - Recirculation Pump Trip (RPT).

Date of issuance: January 26, 1989

Effective date: January 26, 1989, and shall be implemented within 60 days

Amendments Nos.: 164, 161, 135

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

Date of initial notice in **Federal Register:** November 30, 1988 (53 FR 48336). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 26, 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 No. 50-327, Sequoyah Nuclear Plant, Unit 1, Hamilton County, Tennessee

Date of application for amendments: September 21, 1988 as supplemented by letter dated October 25, 1988 (TS 88-28)

Brief description of amendments: The amendment modifies the Sequoyah Nuclear Plant, Unit 1 Technical Specifications. The change revises the limiting condition for operation 3.2.2 and surveillance requirement 4.2.2 to reflect a reduction in the heat flux hot channel factor limit from 2.237 to 2.15. The limit shall be 2.15 instead of 2.237 until an analysis in conformance with 10 CFR 50.46, using plant operating conditions and showing that a limit of 2.237 satisfies the requirements of 10 CFR 50.46(b), has been completed and submitted to NRC.

Date of issuance: January 23, 1989

Effective date: January 23, 1989

Amendment No.: 95

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

Date of initial notice in **Federal Register:** October 5, 1988 (53 FR 39178). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 23, 1989

No significant hazards consideration comments received: No

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

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: September 30, 1988

Brief description of amendments: The amendments allow an increase in the steam generator tube plugging from 7 percent and 15 percent to 18 percent. Also, the maximum FQ limit is increased from 2.15 to a value of 2.19.

Date of issuance: January 17, 1989

Effective date: January 17, 1989

Amendment Nos.: 114 and 97

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

Date of initial notice in **Federal Register**: November 16, 1988 (53 FR 46161). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 17, 1989.

No significant hazards consideration comments received: No.

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

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 March 10, 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).

Indiana Michigan Power Company, Docket No. 50-316, Donald C. Cook Nuclear Plant, Unit 2, Berrien County, Michigan

Date of application for amendment: October 14, 1988

Brief description of amendment: The amendment increases the shutdown margin requirements for operational Modes 4 and 5. The revised requirements are based on an analysis of a potential boron dilution transient.

Date of issuance: January 13, 1989

Effective date: January 13, 1989

Amendment No.: 106

Facility Operating License No. DPR-74. Amendment revises the Technical Specifications.

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

Comments received: No. The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated January 13, 1989

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

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

NRC Project Director: Theodore Quay, Acting.

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

Date of application for amendment: December 21, 1988

Brief description of amendment: The amendment revises testing requirements for the 4.16 KV emergency bus under voltage trip functions set forth in WNP-2 Technical Specification Tables 3.3.3-1 and 4.3.3.1-1. The monthly functional channel test for degraded voltage protection of the Division 1 and 2 buses will include the sensor and its associated 5 second delay relay but will no longer include the secondary 3 second delay relays. The Division 3 protection system will be tested at an interval not to exceed 18 months instead of monthly.

Date of issuance: January 6, 1989

Effective date: January 6, 1989

Amendment No.: 64

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

Public Comment requested as to proposed no significant hazards consideration: No. The Commission's related evaluation of the amendment, finding of emergency circumstances, consultation with the State of Washington, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated January 6, 1989.

Attorneys for licensee: Nicholas S. Reynolds, Esq., Bishop, Cook, Purcell Reynolds, 1400 L Street, NW., Washington, DC 20005-3502 and Mr. G. E. Doupe, Esq., Washington Public Power Supply System, P.O. Box 968, 3000 George Washington Way, Richland, Washington 99352.

Local Public Document Room location: Richland City Library, Swift and Northgate Streets, Richland, Washington 99352.

NRC Project Director: George W. Knighton

Dated at Rockville, Maryland, this 2nd day of February, 1989.

For the Nuclear Regulatory Commission

Gus C. Lainas,

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

[Doc. 89-2837 Filed 2-7-89; 8:45 am]

BILLING CODE 7590-01-D

[Docket No. 50-155, License No. DPR-06, EA 87-80]

Consumers Power Co., Big Rock Point Nuclear Plant; Order Imposing Civil Monetary Penalty

I

Consumers Power Company (licensee) is the holder of Operating License No. DPR-06 issued by the Nuclear Regulatory Commission (NRC/Commission) on August 30, 1962. The license authorizes the licensee to operate the Big Rock Point Nuclear Plant, in accordance with the conditions specified therein.

II

A special safety inspection of the licensee's activities was conducted during the period September 15–19, 1986. The results of the inspection indicated that the licensee had not conducted its activities in full compliance with NRC requirements. A written Notice of Violation and Proposed Imposition of Civil Penalty (Notice) was served upon the licensee by letter dated September 22, 1988. The Notice stated the nature of the violation, the provisions of the NRC's requirements that the licensee had violated, and the amount of the civil penalty proposed for the violation. The licensee responded to the Notice by two letters dated December 1, 1988. In its response, the licensee admitted the facts stated in the violation, but argued that the guidance of the NRC's Modified Enforcement Policy was unduly punitive and not equitably applied when the specifics of the Big Rock Point situation, the complexity of the issues and the size of the plant are considered. The licensee requested that the Commission reconsider the amount of the proposed fine.

III

After consideration of the licensee's response and the statements of fact, explanation, and argument for mitigation contained therein, the Deputy Executive Director for Regional Operations has determined, as set forth in the Appendix to this Order, that the penalty proposed for the violation designated in the Notice of Violation and Proposed Imposition of Civil Penalty should be imposed.

IV

In view of the foregoing and pursuant to section 234 of the Atomic Energy Act of 1954, as amended (Act), 42 U.S.C. 2282, and 10 CFR 2.205, it is hereby ordered that:

The licensee pay a civil monetary penalty in the amount of One Hundred and Eighty-Seven Thousand Five Hundred Dollars ($187,500) within 30 days of the date of this Order by check, draft, or money order, payable to the Treasurer of the United States and mailed to the Director of Enforcement, U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555.

V

The licensee may request a hearing within 30 days of the date of this Order. A request for a hearing should be clearly marked as a "Request for an Enforcement Hearing" and should be addressed to the Director of Enforcement, U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, DC 20555, with a copy to the Regional Administrator, Region III, 799 Roosevelt Road, Glen Ellyn, Illinois, 60137, and a copy to the NRC Resident Inspector, Big Rock Point Nuclear Plant.

If a hearing is requested, the Commission will issue an Order designating the time and place of the hearing. If the licensee fails to request a hearing within 30 days of the date of this Order, the provisions of this Order shall be effective without further proceedings. If payment has not been made at that time, the matter may be referred to the Attorney General for collection.

In the event the licensee requests a hearing as provided above, the issue to be considered at such hearing shall be: whether the proposed civil penalty should be imposed in whole or in part.

For the Nuclear Regulatory Commission.

James M. Taylor,

Deputy Executive Director for Regional Operations.

Dated at Rockville, Maryland this 31st day of January 1989.

Appendix—Evaluation and Conclusion

On December 1, 1988, Consumers Power Company (licensee) replied in two letters to the NRC's September 22, 1988, Notice of Violation and Proposed Imposition of Civil Penalty (notice) regarding environmental qualification (EQ) of electrical equipment admitting that the facts stated in the violations are substantially correct, but raising objections to the NRC's conclusions that a civil penalty was warranted. The licensee states that the deficiencies in the Notice were identified and discussed with the NRC prior to the deadline of November 30, 1985 for compliance with 10 CFR 50.49 and that required corrective action was implemented. In addition, the licensee contends that the amount of the proposed civil penalty is excessive for the significance of the deficiencies and the size of the facility and requests that the Commission reconsider the amount of the proposed fine. The violation is restated below followed by a summary of the licensee's response and the NRC's evaluation and the conclusion.

1. Restatement of Violation

10 CFR 50.49(f) requires each item of electrical equipment important to safety be environmentally qualified by testing and/or analysis.

10 CFR 50.49(k) specifies that requalification of electric equipment important to safety is not required if the Commission has previously required qualification in accordance with "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," November 1979 CDOR Guidelines).

DOR Guidelines, Section 5.2.2, states that type tests should only be considered valid for equipment identical in design and material construction to the test specimen and any deviations should be evaluated as part of the qualification documentations.

Contrary to the above, Consumers Power Company failed to qualify equipment important to safety by appropriate testing and/or analysis as evidence by the following examples:

a. Limitorque Motor Actuator MO-7068, an item of electrical equipment important to safety, was removed from service after 13 years of operation and was subjected to a Loss of Coolant Accident (LOCA) test on April 23, 1975. This actuator was then reinstalled and returned to service in the containment spray system without being qualified by testing and/or analysis to evaluate aging and degradation due to the LOCA test. This condition existed from November 30, 1985 until February 13, 1987, at which time Limitorque Motor Actuator MO-7068 was replaced.

b. Butyl rubber and polyethylene insulated cables, items of electrical equipment important to safety, which had not been environmentally qualified by testing and/or analysis, were installed in various Class 1E circuits inside containment. This condition existed from November 30, 1985 until June 30, 1987, at which time the unqualified cables were replaced.

2. Summary of Licensee's Response

The licensee admits that the facts stated in the violation are substantially correct. However, Consumers Power Company claims that, prior to the EQ deadline, the qualification concerns had been identified and discussed with the NRC and that the licensee had implemented actions to satisfy the concerns. Since the NRC had not notified the licensee to the contrary, the licensee had assumed the concerns had been satisfactorily addressed and its equipment was qualified.

Consumers Power Company also argues that a fine of the magnitude proposed is unreasonable for a generating plant the size and age of Big Rock Point. In fact, on a per megawatt basis, the licensee argues that is the largest fine the Commission has ever proposed for a licensee. The licensee also argues that the safety significance of the examples in the Notice do not warrant a fine in the amount proposed. In summary, the licensee states that, due to the circumstances that apply to the specifics of the Big Rock Point situation and the complexity of the issues involved, the guidance of the modified Enforcement Policy is unduly punitive and has not been equitably applied. The licensee contends that the amount of the proposed civil penalty is excessive and requests that the Commission reconsider the amount of the proposed fine.

3. NRC Evaluation of Licensee's Response

The NRC staff believes the licensee had no reasonable basis for assuming that the NRC had approved its actions to satisfy the identified EQ concerns. As evidenced in various NRC documents, the NRC did identify the document the deficiencies stated in the Notice prior to the EQ deadline (as early as 1983) and in each case identified the need for replacement or new testing and analysis of the unqualified equipment. The licensee's corrective actions were not presented to the NRC until the September 1986 Region III EQ Inspection. During this inspection, the NRC again informed the licensee that the actuator and cables in question were unqualified. The licensee took an unreasonable length of time to correct the identified deficiencies and numerous meetings had to be held between the NRC and the licensee to prompt the licensee and ensure that it took adequate corrective action.

With regard to Limitorque Actuator MO-7068, the licensee claims that the NRC and its consultant, Franklin Research Institute, were aware that Actuator MO-7068 had been tested under LOCA conditions and returned to service after being inspected and refurbished "where needed." Since the NRC had raised no further concerns, the licensee assumed the actuator was qualified for intended service.

The Franklin Research Center Technical Evaluation Report (TER), February 18, 1983, Page 3A, identified Actuator MO-7068 as Category IB, "Equipment Qualification Pending Modification." The summary section of the TER identified the corrective action as "Replace or Rebuild and Qualify."

In its conclusion, the TER stated, "radiation and thermal aging qualification testing has not been performed for this type actuator." The TER also stated this conclusion for other type Limitorque Model SMA-00 actuators.

In the discussion, the TER acknowledged that the Actuator MO-7068 had at one time been subjected by the licensee to a LOCA. However, that test was considered an adequate basis only for interim operation for Type SMA-00 actuators until they were replaced or rebuilt. Further, the TER did not state or imply that the LOCA-tested actuator, i.e., MO-7068, could be returned to service without refurbishment of degraded parts. The licensee, however, returned the actuator to service after the LOCA test without any evidence of refurbishing EQ-related components. Neither the NRC nor Franklin was aware that the actuator had been returned to service without the refurbishing of degraded parts. Based on these considerations, the licensee's claim that the actuator was qualified based on lack of NRC notification to the contrary is not supported.

With regard to the Polyethylene and Butyl Rubber insulated cables, the licensee claims that, in lieu of LOCA-testing the cables, it purchased a test report for $50,000 and qualified the cables by similarity. The licensee assumed the cables were qualified since the NRC had raised no further concern.

The February 18, 1983 Franklin TER identified Polyethylene and Butyl rubber cables as those for which equipment qualification had not been established. In June 1984, NRR identified these cables as unqualified during an EQ inspection. On July 25, 1984, the NRC granted the licensee an extension on the schedule for qualification of these cables until March 31, 1985. Finally, in September 1986, the Region III inspectors identified those cables as unqualified and required replacement or qualification by testing. Despite all these notifications, the licensee did not take timely corrective action. During an April 13, 1987 meeting between the Consumers Power Company and NRC staffs, the licensee committed to replace all Polyethylene and Butyl rubber cable in question. This commitment was documented in an April 15, 1987 Confirmatory Action Letter issued to the licensee by the NRC Region III office.

The licensee claims it spent $50,000 to purchase test reports of similar cables because 10 CFR 50.49 permits qualification by similarity. The licensee claims the NRC was aware of its approach to qualify by similarity and had raised no concerns. The NRC agrees that a licensee may qualify equipment by similarity as this clearly allowed in the regulations. However, when the NRC inspection was conducted, the tests discussed in the purchased reports were found to be deficient in that they did not test similar or identical cable. The NRC had not reviewed the adequacy of these reports until the Region III inspection, at which time the reports were found clearly inadequate for applications at Big Rock, for the reasons given in the Notice.

The licensee claims the NRC SER of November 15, 1985, further confirmed the qualification of these cables because there were no remarks to the contrary. The NRC SER, however, only addressed the approval of the licensee's general approach to resolving outstanding EQ deficiencies, not the adequacy of the resolution of each specific issue. The corrective actions were scheduled to be reviewed during the NRC Region III inspection. Based on the above consideration, the licensee's claim that the cables were qualified by similarity based on lack of NRC notification to the contrary, is not supported.

With regard to the licensee's argument concerning the safety significance of the violation, the NRC staff, under the Modified EQ Policy Enforcement Policy, considers violations of EQ requirements to be safety significant because the electrical equipment required to be qualified are those which are important to safety. This is a case in which it appears that the components were properly categorized as important to safety. If the licensee cannot demonstrate that such components are qualified, for enforcement purposes, a significant violation has occurred. The only exceptions to this practice include those cases in which a documentation deficiency of a minor nature exists which is readily correctable. In this case, the licensee failed to have adequate documentation and would have needed to develop extensive additional information to demonstrate qualification. Therefore, the NRC staff concluded a significant violation existed.

While Consumers Power Company does operate a small reactor, Big Rock Point's size alone is not a sufficient justification for mitigation of a civil penalty. The facility is categorized as a commercial power reactor and as such is subject under the Modified EQ Enforcement Policy, as under the "General Statement of Policy and Procedure for NRC Enforcement Action", 10 CFR Part 2 Appendix C, to the same base civil penalty as all other commercial power reactors. The NRC carefully considered whether it would be advisable to assess lesser civil penalties for smaller commercial power reactors and it was concluded that the inherent risks associated with any size commercial nuclear plant are such that a significant deterrent is needed to motivate a licensee to implement and maintain programs for detection and correction of problems that may constitute or lead to violations of regulatory requirements.

For these reasons, the NRC has concluded that mitigation of the civil penalty is not warranted.

3. Conclusion

The NRC has concluded that this violation occurred as stated and there is no adequate basis for withdrawing the violation or reducing the amount of the civil penalty. Consequently, the proposed civil penalty in the amount of $187,500 should be imposed.

[FR Doc. 89-2977 Filed 2-7-89; 8:45 am]
BILLING CODE 7590-01-M

[Docket No. 50-334]

Duquesne Light Co., Ohio Edison Co., and Pennsylvania Power Co., Beaver Valley Power Station, Unit No. 1; Denial of Amendment to Facility Operating License and Opportunity for Hearing

The U.S. Nuclear Regulatory Commission (the Commission) has denied a request by Duquesne Light Company, (licensee) an amendent to Facility Operating License No. DPR-66, issued to the licensee for operation of the Beaver Valley Power Station, Unit No. 1, located in Beaver County, Pennsylvania. Notice of Consideration of Issuance of this amendment was published in the **Federal Register** on July 15, 1987 (52 FR 26586).

The purpose of the licensee's amendment request was to revise the Technical Specifications (TS) to clarify certain requirements concerning reactor coolant system boron dilution.

The licensee has informed the staff that a revised request will be submitted to address the staff's concerns. The revised submittal is still outstanding. Therefore, the staff decides to deny the amendment request in order to conserve staff resources. This denial will not constitute a prejudice against the licensee's revised submittal which will be treated as a new request.

The licensee was notified of the Commission's denial of the proposed TS change by a letter dated by March 10, 1989, the licensee may demand a hearing with respect to the denial described above. Any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for hearing or 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.

A copy of any petitions should also be sent to the Office of the General Counsel-Rockville, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to Gerald Charnoff, Esquire and Jay E. Silberg, Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037, attorney for the licensee.

For further details with respect to this action, see (1) the application for amendment dated April 30, 1987, and (2) the Commission's letter to the licensee dated.

These documents are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the B.F. Jones Memorial Library, 663 Franklin Avenue, Aliqquippa, PA 15001. A copy of item (2) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Document Control Desk.

Dated at Rockville, Maryland, this 2nd day of February,1989.

For the Nuclear Regulatory Commission.

Peter D. Tam,

Senior Project Manager, Project Directorate I-4, Division of Reactor Projects O/II, Office of Nuclear Regulation.

[FR Doc. 89-2978 Filed 2-7-89; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-260]

Tennessee Valley Authority; Withdrawal of Application for Amendment to Facility Operating License DPR-52

The United States Nuclear Regulatory Commission (the Commission) has granted the request of the Tennessee Valley Authority (the licensee) to withdraw its August 12, 1988 application for amendment, technical specification (TS) change 249, to Facility Operating License DPR-52 for the Browns Ferry Nuclear Plant, Unit 2 located in Decatur, Alabama. TS-249 will be replaced by a new request for changes.

This amendment would have modified the TS by revising the limiting conditions for operation and the surveillance requirements for equipment required by 10 CFR Part 50, Appendix R safe shutdown.

The Commission issued a Notice of Consideration of Issuance of Amendment published in the **Federal Register** on October 19, 1988 (53 FR 4100). By letter dated January 17, 1989, the licensee withdrew the proposed change regarding Appendix R safe shutdown.

For further details with respect to this action, see the application for amendment dated August 12, 1988 and the licensee's withdrawal dated January 17, 1989. These documents are available for public inspection at the Commission's Public Document Room, 2121 L Street NW., Washington, DC and at the Athens Public Library, South Street, Athens, Alabama.

Dated at Rockville, Maryland this 1st day of February 1989.

For the Nuclear Regulatory Commission.

Suzanne Black,

Assistant Director for Projects, Office of Nuclear Reactor Regulation.

[FR Doc. 89-2979 Filed 2-7-89; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-483]

Union Electric Co., Callaway Nuclear Power Plant; Consideration of Issuance of Amendment to Facility Operating License and Opportunity for Hearing

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. NPF-30, issued to Union Electric Company, for operating of the Callaway Plant located in Callaway County, Missouri.

The amendment would change Technical Specification (TS) 4.9.8.1, 4.9.8.2, and the associated Bases to reduce the required Residual Heat Removal (RHR) system flow rate during Mode 6 operation; change TS 4.4.9.3.2, 4.5.2.d, and the associated Bases to delete the RHR autoclosure interlock function; and change TS 3.5.4 and the associated Bases to allow safety injection pumps to be energized with the head on and with water level not above the top of the reactor vessel flange, in Modes 5 and 6.

Prior to issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

By March 10, 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 request for hearing and a 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 petitoner'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 fifiteen (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 that 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.

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, 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 Indentification Number 3737 and the following message addressed to John N. Hannon: 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 Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037, attorney for the licensee.

Nontimely fillings 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 the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

If a request for hearing is received, the Commission's staff may issue the amendment after it completes its technical review and prior to the completion of any required hearing if it publishes a further notice for public comment of its intent to make a no significant hazards consideration finding in accordance with 10 CFR 50.91 and 50.92.

For further details with respect to this action, see the application for amendment dated January 6, 1989, which is available for public inspection at the Commission's Public Document Room, 2120 L Street, NW., Washington, DC 20555, and at the local public document room, 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.

Dated at Rockville, Maryland, this 31st day of January, 1989.

For the Nuclear Regulatory Commission.

Timothy G. Colburn,

Acting Director, Project Directorate III-3, Division of Reactor Projects-III, IV, V and Special Projects, Office of Nuclear Reactor Regulation.

[FR Doc. 89-2960 Filed 2-7-89; 8:45 am]

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