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ENDRIES,J.M. Niagara Mohawk Power Corp.
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Document Control Branch (Document Control Desk)

SUBJECT: Advises that effective 881115 L Burkhardt appointed executive vice president of nuclear operations.

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NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/301 PLAINFIELD ROAD, SYRACUSE, N.Y. 13212/TELEPHONE (315) 474-1511

November 21, 1988
NMPIL 0324

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Re: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Nine Mile Point Unit 2
Docket No. 50-410
NPF-69

Gentlemen:

Niagara Mohawk Power Corporation has recently made a change to its management organization. Effective November 15, 1988, Mr. Lawrence Burkhardt III has been appointed Executive Vice President, Nuclear Operations. Mr. Burkhardt will be responsible for the full range of nuclear issues and activities for both nuclear power plants.

Correspondence previously addressed to Mr. C. V. Mangan, Senior Vice President, should be mailed to Mr. Burkhardt at the following address: Niagara Mohawk Power Corporation, 301 Plainfield Road, Syracuse, New York 13212.

Very truly yours,

NIAGARA MOHAWK POWER CORPORATION

John M. Endries
President

JMM/pns
6032G

xc: Regional Administrator, Region I
Mr. R. A. Capra, Director
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November 21, 1988

DOCKET NO(S). 50-220
Mr. Charles V. Mangan
Senior Vice President
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

SUBJECT: NIAGARA MOHAWK POWER CORPORATION
Nine Mile Point Nuclear Station, Unit No. 1

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 11/16/88 [~~xxxxxxxxxxx~~] ~~comments by 12/16~~.
- [ ] 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

OFFICE	PDI-1						
SURNAME	CVogan						
DATE	11/21/88						

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MA 4/01

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 October 26, 1988 through November 4, 1988. The last biweekly notice was published on November 2, 1988 (53 FR 44247).

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 December 16, 1988 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 Nos. 50-313 and 50-368, Arkansas Nuclear One, Units 1 and 2, Pope County, Arkansas

Date of amendment requests: May 27, 1988

Description of amendment requests: The amendments would modify the Technical Specifications (TSs) for each unit by adding operability and surveillance requirements for the core-exit thermocouples (CETs). The CET system is one of the inadequate core cooling (ICC) monitoring systems. These systems and the associated TSs are required by NUREG-0737, Section II.F.2, as specified by Generic Letter 83-37.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance for the application of criteria for no significant hazards consideration determination by providing examples of amendments that are considered not likely to involve significant hazards considerations (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 new operability and surveillance requirements for the CET system of each unit constitute additional limitations, restrictions, and controls not presently included in the Technical Specifications. Therefore, the proposed amendments are within the scope of the example.

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

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

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

NRC Project Director: Jose A. Calvo

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

Date of application for amendment: May 27, 1988

Description of amendment request: The proposed amendment would change the reactor water level setpoint for the isolation of the Group 1 primary containment isolation valves from low level 2 to low level 3. The proposed amendment also reflects plant modifications that are necessary. New slave units will be added for the low level 3 instrumentation and their tag numbers will be identified in the Technical Specifications. Master trip units will be upgraded and the Technical Specifications will reflect the new tag numbers for low level 2.

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 evaluated the proposed changes in accordance with the standards in 10 CFR 50.92(c) and provided the following analysis:

1. The setpoint change has been evaluated with respect to several operating parameters, including the minimum critical power ratio (MCPR), peak vessel pressure, radiation release, and shutdown capability during abnormal operating transients. Fuel cladding integrity during a loss-of-coolant accident (LOCA) and the reactor response during an anticipated transient without scram (ATWS) event were also evaluated. Results of this evaluation are provided in the GE Topical Report NEDC-30601-P, "Safety Review of Water Level Setpoint Change for Brunswick Steam Electric Plant, Units 1 and 2." As

stated in Section 4.2.3 and 4.2.4 of that report, the change will not cause a reduction in MCPR, an increase in the peak pressure, an increase in radiation release, a cause of equipment damage, a reduction in plant shutdown capability, or a decrease in core cooling capability. The main steam isolation valve (MSIV) water level setpoint change has no impact on LOCA events previously evaluated, nor does it cause consequences of accidents previously evaluated to be increased.

2. Several operating parameters have been evaluated to support the setpoint change, including MCPR, peak vessel pressure, radiation release, and shutdown capability during abnormal operating transients. Fuel cladding integrity during a LOCA and reactor response during an ATWS event were also evaluated. Results of this evaluation are provided in the GE Topical Report NEDC-30601-P, "Safety Review of Water Level Setpoint Change for Brunswick Steam Electric Plant, Units 1 and 2." None of these evaluations indicated that any new or different type of accident would be created by the change. In addition, the present function and structure of the Group 1 isolation valves remains unchanged, thereby eliminating possible operator confusion and training problems that could lead to a new or different type of accident. Therefore, the proposed change does not create the possibility of a new or different kind of accident.

3. The effects of the setpoint change for LOCA events has been reviewed, and it has been determined that the change has no impact. As stated in NEDC-30601-P, large and intermediate LOCA events will not be affected because the rapid depressurization and rapid inventory loss will cause the MSIV to close almost immediately after the accident, before any fuel failure could occur. Thus, the lower MSIV trip will not increase inventory loss from the reactor core or radiation release to the environment. For a small break LOCA, the highest peak cladding temperature for the worst case single failure (i.e., failure of the high pressure coolant injection (HPCI) system) is considerably less than the 2200° F peak clad temperature limit. Therefore, the setpoint change will have no effect on the limiting maximum average planar linear heat generation rate (MAPLHGR).

For a loss of feedwater flow event under the proposed amendment, the reactor would not be isolated while HPCI and reactor core isolation [cooling] (RCIC) are operating. Reactor core isolation cooling system flow would compensate for steam flow through the turbine control valves to the main condenser, thereby maintaining water level above low level 3, keeping the MSIVs open, and preventing the S/RVs from opening. Thus, the MSIV setpoint change will not compromise core cooling capability for the loss of feedwater flow event. Furthermore, it reduces suppression pool heatup for this event because the main condenser is available for a longer time.

The low level 3 reactor water level setpoint for the Group 1 primary containment isolation system valves still "ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents" as demonstrated by the evaluation in Sections 4 and 5 of NEDC-30601-P. Thus, for the reasons described above, the margin of safety is not reduced and may actually be increased.

Based on the above, the licensee has determined that the proposed amendment does not involve a significant hazards consideration. The NRC staff has reviewed the licensee's no significant hazards consideration determination and agrees with the licensee's analysis. Accordingly, the Commission proposes to determine that the requested amendment does not involve a significant hazards consideration.

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

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

NRC Project Director: Elinor G. Adensam

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

Date of application for amendment: June 9, 1988

Description of amendment request: The proposed change to Brunswick Steam Electric Plant, Unit No. 1, (BSEP) Technical Specification Tables 3.3.5.2-1 and 4.3.5.2-1 is requested to address alternate shutdown capability requirements associated with 10 CFR Part 50, Appendix R.

Reactor vessel water level transmitter B21-LT-N026A currently provides indication on the remote shutdown panel (B21-LI-R604AX), as well as on the control panel in the control room (via B21-LT-604A), and level transmitter B21-LT-N026B provides indication only on the control panel (via B21-LI-R604B). The remote shutdown panel is located in the reactor building. The proposed modification would have level transmitter B21-LT-N026A providing indication to only the control panel in the control room (on B21-LI-R604A) and level transmitter B21-LT-N026B will provide indication on the remote shutdown panel (B21-LI-R604BX) and, via the remote shutdown panel, on the control panel (B21-LI-R604B).

The licensee states that these modifications are being made to address alternate shutdown capability requirements associated with 10 CFR Part 50, Appendix R, Section III.G. The above described level transmitters and indicators are identified in the Technical Specifications and, subsequently, need to be changed to support the modification.

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.

Carolina Power & Light Company (the Company) has reviewed this proposed license amendment request and determined that its adoption would involve no significant hazards consideration for the following reasons:

1. The instrumentation being rewired provides reactor water level indication as part of the plant monitoring instrumentation required for 10 CFR Part 50, Appendix, R, Section III.G. It provides no direct protection against any of the accidents identified in Chapter 15 of the Updated Final Safety Analysis (UFSAR). By rewiring these instruments, the Train A instrumentation will feed the control room and the Train B instrumentation will feed the remote shutdown panel thereby satisfying the requirements of 10 CFR Part 50, Appendix R, Section III.G. The purpose and function of the instrumentation will not change; only the indication point will be exchanged between instrument trains. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Rewiring of level transmitter loops B21-LT-N026A and B21-LT-N026B will allow the Company to safely shutdown the unit using the Alternate Safe Shutdown Procedures. The possibility of a new or different kind of accident from any accident previously evaluated will not be created because this instrumentation will not be performing any different function from its current function. This modification is being made to address the commitments associated with 10 CFR Part 50, Appendix R, Section III.G which require the Train A instrumentation to feed the control room and the Train B instrumentation to feed the remote shutdown panel. The new configuration will ensure consistent indication, i.e., Train A or Train B, in both the control room and at the remote shutdown panel. The necessary indication will be available to the operator at the proper location under fire scenarios which would take out either the Train A or Train B instrumentation.

3. The proposed modification will ensure proper indication in the appropriate area in

the event of a fire that takes out either the Train A or Train B instrumentation. Currently, transmitter B21-LT-N026A feeds the remote shutdown panel, and transmitter B21-LT-N026B feeds the control room. Commitments associated with compliance with 10 CFR Part 50, Appendix R, Section III.G requires that Train A feed the control room and Train B feed the remote shutdown panel. This is to ensure that in the event of a fire that takes out Train A, there will be indication to the remote shutdown panel from Train B, and in a fire that takes out Train B, there will be indication to the control room from Train A. Thus, the proposed amendment does not involve a significant reduction in the margin of safety.

Based on the above reasoning, the licensee has determined that the proposed changes involve no significant hazards consideration. The NRC staff has reviewed the licensee's no significant hazards consideration determination and agrees with the licensee's analysis. Accordingly, the Commission proposes to determine that the requested amendment does not involve a significant hazards consideration.

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

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

NRC Project Director: Elinor G. Adensam

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

Date of application for amendment: June 27, 1988

Description of amendment request: The proposed amendment would change Technical Specification (TS) Tables 3.3.5.6-1, 3.3.5.6-2 and 4.3.5.6-1 to replace instrument tag number TS-CR-863 with TS-CIT-863-3. This change is needed as a result of planned upgrading of instrumentation during the Brunswick Steam Electric Plant, Unit 1, refueling outage of November 1988. Item 2 in each of the above tables lists chloride leak detection instrumentation in the condensate pump discharge. This instrumentation provides indication of chloride intrusion in the feedwater and condensate systems. Chlorides pose a long-term threat to the integrity of stainless steel piping systems. The change is necessary due to a plant modification that will replace the instrument represented by TS-CR-863-3 with an upgraded conductivity cell and analyzer represented by tag number TS-CIT-863-3. The upgraded components are capable of detecting and compensating for temperature transients that may occur in the sample being analyzed. The new conductivity analyzer will provide a direct and continuous reading without relying on a recorder, and will also provide an output to a recorder for trending purposes.

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 determined that:

1. The accidents analyzed in Chapter 15 of the Updated FSAR are not affected by the chloride leak detection instrumentation change because the function of the instrument is not altered and the chloride limits established in TS 3/4.4.4 are not being changed. In addition, the new instruments being installed are capable of detecting and compensating for temperature transients which may occur in the sample being analyzed. The current system requires additional data processing to achieve the same results. Based on this reasoning, CP&L has determined that the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. As stated above, the chloride leak detection instrumentation provides protection from long-term piping degradation in the feedwater and condensate systems caused by chloride intrusion. No possibility of a new or different kind of accident is created because the new instruments perform the same basic function as the ones they are replacing. Also, the reactor coolant system chloride limits established in TS 3/4.4.4 are not being changed. The new instrument has enhanced capabilities: it processes the data into a more useful form prior to readout. Based on the above reasoning, CP&L has determined that the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The margin of safety is not reduced because, as stated above, the new instruments perform the same basic function as the ones they are replacing and the chloride limits established in TS 3/4.4.4 are not being changed. In fact, the new instruments have enhanced capabilities which may provide the user with better data, thereby providing earlier indication of chloride intrusion, and perhaps avoiding long-term problems with pipe degradation due to chloride intrusion. Based on this reasoning, CP&L has determined that the proposed amendment does not involve a significant reduction in the margin of safety.

Based on the above reasoning, the licensee has determined that the proposed changes involve no significant hazards consideration. The NRC staff has reviewed the licensee's no significant hazards consideration determination and agrees with the licensee's analysis. Accordingly, the Commission proposes to determine that the requested amendment does not involve a significant hazards consideration.

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

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

NRC Project Director: Elinor G. Adensam

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

Date of application for amendments: November 26, 1986 and January 14, 1988

Description of amendments request: In accordance with the requirements of 10 CFR 73.55, the licensee submitted an amendment to the Braidwood Nuclear Power Station Security Plan to reflect recent changes to that regulation. The proposed amendment would modify paragraph 2.F of Facility Operating License Nos. NPF-72 and NPF-77 to require compliance with the revised plan.

Basis for proposed no significant hazards consideration determination: On August 4, 1986 (51 FR 27817 and 27822), the Nuclear Regulatory Commission amended Part 73 of its regulations, "Physical Protection and Plants and Materials," to clarify plant security requirements to afford and increased assurance of plant safety. The amended regulations required that each nuclear power reactor licensee submit proposed amendments to its security plan to implement the revised provisions of 10 CFR 73.55. The licensee submitted its revised plan on November 26, 1986 and January 14, 1988, to satisfy the requirements of the amended regulations. The Commission proposes to amend the license to reference the revised plan.

In the Supplementary Materials accompanying the amended regulations, the Commission indicated that it was amending its regulations "to provide a more safety conscious safeguards system while maintaining the current levels of protection" and that the "Commission believes that the clarification and refinement of requirements as reflected in these amendments are appropriate because they afford an increased assurance of plant safety."

The Commission has provided guidance concerning the application of the criteria for determining whether a significant hazards consideration exists by providing certain examples of actions involving no significant hazards considerations and examples of actions involving significant hazards consideration (51 FR 7750). One of these examples of actions involving no significant hazards considerations is example (vii) "a change to conform a license to changes in the regulations, where the license change results in very minor changes to facility operations clearly in keeping with the regulations." The changes in this case fall within the scope of the example. For the foregoing reasons, the Commission proposes to determine that the proposed amendment involves no significant hazards consideration.

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

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

NRC Project Director: Daniel R. Muller

Commonwealth Edison Company, Docket No. 50-374, LaSalle County Station, Unit 2, LaSalle County, Illinois

Date of application for amendment: September 14, 1988

Brief description of amendment: The proposed amendment to Operating License No. NPF-18 would revise the LaSalle Unit 2 Technical Specifications in support of the second reload for LaSalle Unit 2. Startup for Cycle 3 is currently scheduled for January 1989. The proposed reload fuel and analyses including the previously approved SAFER/GESTR-LOCA Loss-of-Coolant Accident (LOCA) Analysis are changes resulting from analyses performed to expand the operating region and allow equipment out-of-service and changes that are administrative or provide clarification. The proposed changes for LaSalle Unit 2 are identical to those previously submitted and approved for use at LaSalle Unit 1, except for minor calculation differences in the results for transient analyses and include:

1. Provision for operation in the expanded operating domain including revised APRM and RBM setpoint changes incorporated using standard and previously approved methodology.

2. Use of extended burnup fuel (GE 8x8EB) with increased LHGR limit of 14.4 Kw/ft.

3. Use of improved transient and LOCA analysis methods which allow use of a lower tau-B value in determining the MCPR operating limit as a function of scram time, and deletion of the single loop MAPLHGR limit multiplier of 0.85.

4. Provision for operation with certain equipment inoperable or out of service. Specifically, one of the following systems or components may be out of service when the appropriate Technical Specification Actions are satisfied:

a. Turbine Bypass System

b. End-of-Cycle Recirculation Pump Trip (EOC-RPT)

c. One Safety Relief Valve (SRV)

d. Feedwater Heaters

5. Several changes for clarification or administrative purposes were proposed including:

a. Deletion of GEXL correlation and GETAB statistical model in the bases of the safety limit section.

b. Revision to the Control Rod Program Controls Technical Specification to require the RWM to be demonstrated operable in Operational Condition 1, prior to reaching 20% power, when reducing thermal power.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether no significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from an accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The licensee has determined, and the NRC staff agrees, that the proposed amendment will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated because the use of the proposed operating limits is specifically analyzed to ensure that the input assumptions of all existing transient and accident analyses remain valid. These analyses are performed using a methodology which has received review and approval for other similar plants including LaSalle Unit 1. The Technical Specification Actions included in the proposed revisions do not significantly affect the probability of an accident previously analyzed because the required time intervals for corrective action are consistent with the existing specifications.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed MCPR, MAPLHGR, and LHGR limits represent limitations on reactor operating state which do not directly affect the operation, or function of any system or component. As a result, there is no impact on or addition of any systems or equipment whose failure could initiate an accident. The proposed operating domain is evaluated to retain the originally required design margins to system integrity during normal operation, transients and accidents and therefore, do not cause significant new loads or stresses on mechanical systems or boundaries. The proposed allowances for operating with prescribed equipment inoperable or out-of-service do not cause physical changes to any systems and therefore do not induce new failure modes.

3. Involve a significant reduction in the margin of safety because no changes to safety limits protective system logic or design are involved. The analyses used to evaluate reactor and system performance are performed using standard methods and the calculated operating limits maintain conservative margin to safety limits to accommodate the anticipated performance during transients and accidents. Changes which are administrative in nature do not affect the operating limits of the plant or the consequences of analyzed transients.

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

Attorney to licensee: Michael Miller, Esq., Sidley and Austin, One First National Plaza, Chicago, Illinois 60603

NRC Project Director: Daniel R. Muller

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

Date of application for amendments: July 7, 1988

Description of amendments request: There are two changes associated with the proposed license amendment. The first change results from the completion

of a Unit 2 Detailed Control Room Design Review (DCRDR) Human Factors modification which resulted in the relocation of the drywell temperature indicator from the 902-21 (back) panel to the 902-3 (front) panel. Such a change would be incorporated into Technical Specifications (TS) Table 3.2-4 and 4.2-2 of DPR-30. The remaining proposed changes would correct typographical errors associated with the same TS tables.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists. As stated in 10 CFR 50.92(c), a proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (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. Pursuant to 10 CFR 50.91(a) the licensee has provided the following evaluation of their amendment application addressing these three standards.

CECo has evaluated the proposed Technical Specifications changes and determined that they do not present a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92(c), operation of QCNPS in accordance with the proposed changes:

(1) Will not involve a significant increase in the probability or consequences of an accident previously evaluated because only the location of the drywell temperature indication has been changed (from the back to the front panel) in the Control Room. This is an enhancement over the previous location to make it more observable for operators. Functions and range of the drywell instrument remain the same. This modification was considered to be a change in the conservative direction.

(2) Will not create the possibility of a new or different kind of accident from any accident previously evaluated because there were no hardware changes (addition or deletion of equipment) per-se, nor are there any new modes of operation associated with this amendment. The changes to Tables 3.2-4 and 4.2-4 reflect changes to equipment (instrumentation) location only.

(3) Will not involve a significant reduction in the margin of safety because revising an instrument location readout in the control room does not adversely affect the operation of any plant systems. Therefore, the margin of safety has not been unchanged as a result of this change.

The NRC staff has reviewed the licensee's evaluation related to the proposed changes and concurs with their conclusions.

In addition, administrative and editorial TS changes are considered representative of example (i) in the Commission's guidance (51 FR 7751) for examples of no significant hazards, which is defined as "a purely administrative change to TS; for example a change to achieve consistency throughout the Technical Specification, correction of an error, or change in nomenclature."

Therefore the NRC staff proposes to determine that this amendment request does not involve significant hazards considerations based upon a preliminary review of the application, the licensee's evaluation of no significant hazards, and NRC guidance.

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

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

NRC Project Director: Daniel R. Muller

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: October 5, 1988

Description of amendments request: This amendment would delete Figure 6.1-1, 'Corporate Organization," and Figure 6.1-2, "Station Organization," from the Technical Specifications (TS) and would revise Section 6 to require inclusion of these organization charts in the QA Topical Report. However, the NRC will continue to be notified of licensee organization changes through other regulatory controls. In accordance with 10 CFR 50.34(b)(6)(i), the applicant's organizational structure is required to be included in the Final Safety Analysis Report (FSAR). Chapter 13 of the FSAR provides a description of the station organization and a detailed organization chart. Updates to the FSAR are required by 10 CFR 50.71(e) to be submitted annually to the NRC. Even though Figures 6.1-1 and 6.1-2 would be deleted from TS, Section 6 of the TS would be revised to require inclusion of these organization charts in the CECo QA Topical. Whereupon, Appendix B to 10 CFR Part 50, and 10 CFR 50.4(b)(7), will govern any changes made to the organization as it is described in the Quality Assurance (QA) Program. Finally, it is CECo's normal practice to inform the NRC of organizational changes affecting their nuclear facilities prior to implementation.

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) involve a significant increase in the probability of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. CECo evaluated the proposed TS changes and determined, and the NRC staff agrees that:

(1) The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated because deletion of the organization charts from the TS does not affect plant operation, nor does it involve any physical modification of the plant. Furthermore, the aforementioned administrative and regulatory controls remain in force to ensure that organizational changes are reviewed by the NRC.

(2) The proposed amendment does not create the possibility of a new or different kind of accident than previously evaluated because the proposed change is administrative in nature; and does not physically alter any systems or components, or the way they are operated.

(3) The proposed amendment does not involve a significant reduction in a margin of safety because CECo through its Quality Assurance programs, and its commitment to maintain only qualified personnel in positions of responsibility, and other required controls, assures that safety-related operations will be performed at a high level of competence. Futhermore, this amendment does not change any setpoints or operating parameters. Consequently, removal of organization charts from the Technical Specifications will not affect the margin of safety. The NRC staff has reviewed the licensee's evaluation related to the proposed changes and concurs with their conclusions.

In addition, the associated editorial TS changes proposed by CECo are considered representative of example (i) in the Commission's guidance (51 FR 7751) for examples of no significant hazards, which is defined as "a purely administrative change to TS: for example a change to achieve consistency throughout the Technical Specifications, correction of an error, or change in nomenclature."

Therefore the NRC staff proposes to determine that this amendment request does not involve significant hazards considerations based upon a preliminary review of the application, the licensee's evaluation of no significant hazards, and NRC guidance.

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

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

NRC Project Director: Daniel R. Muller

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: July 26, 1988

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3/4.7.6 to clarify the emergency power requirements for the Control Room Area Ventilation System. The word "train" is substituted for the word "system" where reference is made to one of the two independent trains which comprise the ventilation system for the control room area shared by the two Catawba units. Also, TS 3/4.7.6 and its Bases are revised to eliminate the possibility of misinterpreting the existing TS requirements for diesel generator (D/G) operability when one or both Catawba units are shutdown.

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 July 26, 1988, provided the following discussion and analysis with regard to the three 10 CFR 50.92 standards:

The Control Room Area Ventilation (VC) and Chilled Water (YC) Systems combine to form one system which is designed to maintain a suitable environment in the following plant areas at all times: Control Room, Cable Room, Battery Room, Switchgear Rooms, Motor Control Center Rooms, and the Electrical Penetration Rooms at elevation 594+0. The VC/YC System is shared between both Units. There are two 100% redundant trains of VC/YC equipment. Each is capable of being powered by Unit 1 or Unit 2 Essential Auxiliary Power, but under normal conditions both trains are aligned to Unit 1. Two diesel generators (D/Gs) are provided per Unit to energize the Essential Auxiliary Power buses during emergency conditions.

Technical Specification 3.7.6 specifies that two independent trains of VC/YC shall be Operable during all operational modes. If one train becomes inoperable while either Unit is in Mode 4, Hot Shutdown, or above, the inoperable train must be restored to Operable status within seven days, or the operating Units must be shutdown. If both Units are below Mode 4 and one train is inoperable, the train must be restored to Operable status within seven days or the Operable train must be operated in the filter mode. If both trains are inoperable, or with the Operable train not capable of being powered by an Operable emergency power source, all core alterations and positive reactivity changes must be suspended on both Units. The requirement for an Operable emergency power source is only specifically stated for Units operating below Mode 4. However, the bases for Technical Specification 3.7.6 states that the operability of VC/YC ensures that ambient air temperature does not exceed allowable limits for equipment and instrumentation, and the Control Room will remain habitable, during and following all credible accident conditions. This implies that an Operable emergency power supply should be a prerequisite to VC/YC operability in all modes.

Technical Specification 3.8.1.1 specifies for each individual Unit that two separate and independent D/Gs are required to be Operable per Unit, if the Unit is in Mode 4, or above. Below Mode 4, Technical Specification 3.8.1.2 applies and only one D/G is required Operable per Unit. Action Statement c. for Technical Specification 3.8.1.1 specifies that when one D/G becomes inoperable, all required systems (or trains) that depend on the remaining Operable D/G as a source of emergency power, must be verified Operable within two hours, or the Unit must be shutdown. This is intended to provide assurance that a loss of offsite power event, while one D/G is inoperable, will not result in a complete loss of safety function of critical systems. It is also the reason Technical Specification 3.7.6 does not specifically require that VC/YC have an Operable emergency power source in Mode 4, or above. This action statement is deficient with respect to VC/YC because VC/YC is a Unit shared system, and Technical Specification 3.8.1.1 applies only to individual Units. In order for VC/YC operability to be protected by this Action Statement, the D/G must become inoperable while the Unit is in Mode 4, or above. There is no such Action Statement in Technical Specification 3.8.1.2 since only one D/G is required Operable, and the Unit is already shut down.

This amendment request would remove the ambiguity as to the emergency power source requirements for the VC System by stating the requirements in the VC System Technical Specification.

To clarify the requirements the Specification is to be split into two separate Specifications. The first Specification (Technical Specification 3.7.6.1) would state the requirements for the VC System when either unit is in Modes 1, 2, 3 or 4. This Specification will now specifically require that each train of the VC System be capable of being powered by an Operable emergency power source whenever either unit is in Mode 1, 2, 3 or 4. This will alleviate the confusion which is currently contained in the Specification as to the emergency power source requirements when one unit is in Modes 1, 2, 3 or 4 and the other unit is in Modes 5 or 6.

Technical Specification 3.7.6.2 is being proposed to clearly state the VC System requirements when both units are in Modes 5 or 6.

The proposed changes to the VC System Specification will also specify that the requirements are to be performed on a per train basis. The VC System is comprised of two independent and redundant trains. The current wording is that which appears in the Standard Technical Specification and should be changed to more clearly reflect what is in place at Catawba.

These proposed changes will add clarification to the requirements of the VC System Specification. The changes will not delete any current requirements or operating restrictions contained in the Specifications and will not allow the plant to be operated in any different mode or configuration. As such, these changes are considered to be administrative in nature and do not involve any Significant Hazards Considerations.

The proposed amendment does not involve an increase in the probability or consequences of any previously evaluated accident. The operating parameters and the design of the VC System are unchanged and no new modes of operation will be introduced by this amendment request. All previous accident analyses are still applicable and remain unchanged by this proposal.

The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes will not change the design or allowed modes of operation of the VC System. As such, no new failure modes are introduced and no new types of accidents are possible.

The proposed amendment does not involve a significant reduction in a margin of safety. These changes will add clarification to the

existing Specification without changing any of the current requirements.

For the above reasons, Duke Power concludes that this proposed amendment does not involve any Significant Hazards Considerations.

The staff has reviewed the proposed changes to TS 3/4.7.6, and agrees with the licensee's evaluation of each of these changes with respect to the three standards of 10 CFR 50.92.

On this basis, the Commission has concluded that the requested changes 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: October 6, 1988

Description of amendment request: The proposed amendments would revise Technical Specification (TS) Table 3.3-3, Item 14.g., to reflect a proposed modification to the pumphouse pit level instrumentation of the Nuclear Service Water (RN) System. The system is designed to supply cooling water to various heat loads in both the safety and non-safety portions of each unit. This modification would change the swapover logic of the RN system.

There are currently four level transmitters per pit at the RN pumphouse. Two are safety-related and two are not safety-related. The modification will upgrade 1 out of the 2 non-safety level transmitters per pit to safety grade. This would accommodate a 2 out of 3 logic instead of the present 1 out of 2 logic. Past experience has shown that a single spurious failure to the "low" position of one level transmitter can initiate a swapover when there is an adequate water level in the RN pits. Inadvertently challenging the system with numerous valves changing position and starting all RN pumps is unnecessary and reduces the reliability of the system.

The failure mode of all the safety grade level transmitters is the same. They fail low on loss of power. This is desirable in order to realign suction from Lake Wylie to the Standby Nuclear Service Water Pond (SNSWP) which is the ultimate heat sink.

The proposed amendments would also temporarily waive the requirements of the Action Statement for Item 14.g. in Table 3.3-3, for 48 hours per pit, on a one time basis in order to allow orderly implementation of this modification. During this time at least one RN pit will be available. The 48 hours is needed for implementation of the modification on each pit separately. During this period, the pit will be inoperable only from the standpoint of automatic realignment to the SNSWP from its normal supply if low level is sensed in the affected pit. All necessary automatic functions would still occur in the opposite pit. The only automatic valve actuation which is activated by train specific pit level instrumentation is the loop cross-over isolation valves. Closure of these valves is only required in the event of design basis accident accompanied by a failure of a pit supply valve to open when an emergency diesel generator or nuclear service water pump is out-of-service for extended maintenance. All four diesel generators and nuclear service water pumps will be maintained in an operable status for the duration of the requested 48 hour period. Therefore, the RN system would be capable of performing its design function during any design basis event, including any concurrent postulated single failure, throughout the requested 48 hour period.

In a letter to Duke Power Company dated September 30, 1987, the NRC staff noted that this proposed modification would improve the overall reliability of the RN System.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR Part 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 possiblity of 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 proposed modification would increase the reliability of the RN system by eliminating unnecessary actuations of the swapover instrumentation and components, and during implementation of the modification the system would be capable of performing its intended function during any design basis event.

The proposed amendments do not create the possibility of a new or different kind of accident from any accident previously evaluated because the RN system design basis would not be changed as a result of this modification, and the proposed modification would improve the reliability of the system.

The proposed amendments do not involve a significant reduction in a margin of safety because the modification would enhance the reliability of the RN system by decreasing the likelihood of inadvertent actuations, and during implementation of the modification the RN system would be capable of performing its intended function during any design basis event.

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

Date of amendment request: October 7, 1988

Description of amendment request: The proposed amendment would revise Section 6.2 "Organization" of the Technical Specifications (TS) to delete the offsite and onsite organization charts, Figures 6.2-1 and 6.2-2. In place of the charts, the revision would add general requirements which capture the essential features of the organizational structure that are defined by the existing organization charts.

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 October 7, 1988, provided the following discussion and analysis with regard to the three 10 CFR 50.92 standards:

This proposed amendment would incorporate the guidance contained in the NRC's Generic Letter 88-06, dated March 22, 1988. The Generic Letter provided for deletion of the organization charts contained within Section 6 of the Technical Specifications provided certain statements be added to cover particular administrative control requirements.

This proposed amendment has been developed based on the Generic Letter guidance.

The NRC Staff concluded and Duke Power concurs that the removal of organization charts from the Technical Specifications will provide greater flexibility to implement changes in organization structure but will not reduce plant safety.

The proposed amendment does not involve an increase in the probability or consequences of any previously evaluated accident. This amendment is administrative in nature and does not change the design or operation of the facility. The accident analyses are therefore unaffected by this proposal.

The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The design and operation of the facility will not be changed by this amendment and no new modes of operation will be introduced.

The proposed amendment does not involve a significant reduction in a margin of safety. This change is administrative in nature and therefore has no effect on any margin of safety.

The NRC's discussion in their Generic Letter 88-06 concluded that any facility incorporating the changes outlined in the Generic Letter will have greater flexibility to implement changes to their organizational structures and will not reduce plant safety.

For the above reasons, Duke Power concludes that this proposed amendment does not involve any Significant Hazards Considerations.

The NRC staff has reviewed the licensee's submittal against the guidance provided in Generic Letter (GL) 88-06, "Removal of Organization Charts from Technical Specification Administrative Control Requirements." The proposed TS revisions, deleting the organization charts and adding more flexible provisions regarding organizational structure, are in accord with this guidance. As also recommended in GL 88-06, the Final Safety Analysis Report (FSAR) contains the offsite and onsite organization charts (Figures 13.1.1-1 and 13.1.2-1) and the qualifications for those positions designated by the charts as requiring a Senior Reactor Operator or Reactor Operator license (FSAR, Section 13.1.3.1). The staff therefore finds that the proposed revisions do not adversely affect the organizational characteristics which are important to the safe operation of the facility. The staff also agrees with the licensee's evaluation of the proposed revisions with respect to the three standards of 10 CFR 50.92.

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

Local Public Document Room location: 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

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

Date of amendment request: October 24, 1988

Description of amendment request: The proposed amendment would extend the interval for several 18-month surveillances: reactor trip system response time, reactor trip bypass breakers automatic undervoltage trip check, engineered safety feature (ESF) logic response time, manual actuation switches for several ESF systems, reactor trip P-4 interlock, seismic monitoring instruments, containment isolation check valve lift tests, containment isolation phase B isolation valve actuations, containment recirculation spray valve actuations, diesel generator maintenance inspection, and battery discharge test. To perform these surveillances, the licensee would have to shutdown the unit. Therefore, the licensee requested that the surveillances specified above be permitted to be done at the first refueling outage, but no later than April 1, 1989. If approved, the amendment would permit extension of the 18-month intervals by several days to about three months. The amendment would thus avert a reactor shutdown only to perform surveillances. The licensee agreed to perform these surveillances if there is an unscheduled shutdown of sufficient duration before the refueling outage.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The proposed schedular extensions are not made as a result of, nor would they lead to any design changes. When approved by the staff, only certain surveillance intervals would be extended. The affected systems will continue to perform as stated in the licensee's Final Safety Analysis Report (FSAR); other requirements regarding these components/systems are not changed. From experience in the past, surveillances were performed usually to confirm that components were operable as designed. The probability of finding inoperable components was not high, and consequently, the probability of occurrence of an accident can only be increased by an insignificant amount if the surveillance interval is extended by a small amount. Therefore, the answers to questions (1) and (2) would be negative. The safety limits assumed in FSAR analyses are not affected by the proposed amendment since all assumptions are expected to remain unchanged. Hence the answer to question (3) is also negative.

On such basis, the staff proposes to determine that the requested amendment involves no significant hazards consideration.

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

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

NRC Project Director: John F. Stolz

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

Date of amendment request: October 20, 1988

Description of amendment request: The amendment would change the maximum allowable control element assembly (CEA) drop time from 2.7 seconds to 3.1 seconds. The application is the result of the licensee's review of NRC Information Notice No. 88-47 entitled, "Slower-than-Expected Rod-Drop Time Testing."

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

The licensee addressed the above three standards in the amendment application and made a no significant hazards consideration determination. In regard to the first standard, the licensee provided the following analysis:

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

The change does not affect any active hardware involving plant operation; rather it affects an acceptance criterion for confirming the required performance of the existing control element assembly (CEA) hardware. Therefore, the proposed change does not increase the probability of an accident previously analyzed.

The impact of changing the CEA drop time from 2.7 to 3.1 seconds on all safety analysis related Design Basis Events (DBE's), for which a scram of the CEA's is predicted, was assessed by specifically re-analyzing only the most limiting events with respect to the various safety analysis fuel and system criteria. In particular, the following events were re-analyzed:

Loss of Condenser Vacuum (LOCV)
Loss of Forced Reactor Coolant Flow
Pre-Trip Steam Line Break (SLB)
Hot Full Power CEA Ejection (CEA Ejection)

It has been demonstrated that the events are either totally unrelated to CEA drop time considerations or are not significantly impacted. Additionally, it was demonstrated for each potentially impacted analysis that the consequences of the analysis remain unchanged or are bounded by the existing analysis. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

With respect to the second standard, the licensee stated:

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

The proposed change in the Technical Specifications does not affect any active hardware involving plant operation; rather it affects only an acceptance criterion for confirming the required performance of the existing CEA hardware. Therefore, the proposed change would not create the possibility of a new or different kind of accident from any previously evaluated.

With regard to the third standard, the licensee provided the following rationale:

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

The increased CEA drop time has been evaluated for its impact on the current licensed safety analysis. The results of the re-analysis for those transients which are potentially impacted by the proposed change show that the reference analyses are valid or that the new analysis results still show acceptable results with respect to the acceptance criteria. Therefore, there is no significant reduction in the margin of safety.

The staff has reviewed the analysis provided by the licensee in support of a no significant hazards consideration determination. The staff believes that the licensee has met the standards 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: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 33450

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: October 24, 1988

Description of amendment request: The amendment would expand the Departure from Nucleate Boiling and Linear Heat Rate-related Axial Shape Index limits contained in Figures 3.2-4 and 3.2-2, respectively. In addition, a similar expansion of limits is proposed for the Linear Heat Rate-related Limited Safety System Setpoints as contained in Figure 2.2-2. The licensee is requesting these changes to give the plant greater flexibility at low and intermediate power levels.

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

The licensee addressed the above three standards in the amendment application and provided a no significant hazards consideration determination. In regard to the first standard, the licensee provided the following analysis:

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

The Axial Shape Index (ASI) limits are used as initial assumptions for all Design Basis Events (DBEs) evaluated in the safety analysis. The expansion of these ASI limits for lower powers is applicable only to those DBEs that are evaluated between hot full and hot zero power. Events are not typically analyzed at intermediate power levels. Events initiated from intermediate power levels (100% greater than initial power greater than 0%) are unaffected since these are bounded by the results of events initiated from either the full power or zero power events.

The existing safety analyses for these events use input parameters that are axial shape dependent, such as scram reactivity insertion curves, which are more adverse (conservative) than the Technical Specification Limiting Condition for Operation (LCO) and Limiting Safety System Setpoint (LSSS) axial shape limits at all power levels in order to bound future cycles' operation. It was verified, using current methodology and the proposed ASI limits, that the current safety analysis remains valid.

The current ASI limits allowed by the Departure from Nucleate Boiling (DNB) and Linear Heat Rate (LHR) LCOs and LSSSs are expanded for greater operational flexibility at lower powers. [These] proposed change[s] will not increase the probability or consequences of an accident previously evaluated because the proposed limits are still conservative with respect to the actual calculated limiting values.

With regard to the second standard, the licensee stated:

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

The proposed changes in the Technical Specifications do not affect any active hardware involving plant operation, nor do they alter the assumptions or methodology of the safety analyses. Therefore, they will not create the possibility of a new or different

kind of accident from any previously evaluated.

With regard to the third standard, the licensee provided the following rationale:

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

The wider ASI bands allowed at lower powers have been reviewed for their impact upon the current licensed safety analysis. The licensed safety analysis of record remains unchanged due to the expanded ASI range for low powers. Therefore, there is no significant reduction in a margin of safety.

The staff has reviewed the licensee's no significant hazards consideration determination analysis. Based upon the review, the staff believes that the licensee has met the three standards. Based upon the above discussion, the staff proposes to determine that the proposed changes do not involve a significant hazards consideration.

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

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: October 24, 1988

Description of amendment request: The proposed amendment would relax the maximum allowable primary loop resistance temperature detector (RTD) delay time from 8 seconds to 16 seconds. This delay time is a factor that must be considered in the thermal margin/low pressure reactor trip. According to the licensee, this change would provide increased operational flexibility without decreasing the margin of safety.

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

The licensee addressed the above three standards in the amendment application and provided a no significant hazards consideration determination. In regard to the first standard, the licensee provided the following analysis:

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

The Resistance Temperature Detector (RTD) response time affects only measurement hardware which passively ascertains the coolant temperature condition, not active hardware impacting the plant's physical thermal-hydraulic operations. Therefore, the proposed change does not increase the probability of occurrence of any accident. As described before, the safety analyses demonstrate that the same degree of protection is available at the longer RTD response times since the ex-core power detectors (which do not depend on RTD response time) now provide the required protection when more realistic physics inputs are used. With regard to operations, it should be noted that the plant will be operated in the same manner as before. Therefore, the calculated consequences of the accidents will not increase due to this change.

With regard to the second standard, the licensee stated:

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

The proposed change to the Technical Specifications does not affect any active hardware involving plant operation, nor does it alter the basic methodology of the safety analyses. Therefore, it will not create the possibility of a new or different kind of accident from those accidents previously evaluated.

With regard to the third standard, the licensee provided the following rationale:

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

The value of the RTD response time affects the ability of the delta T-power calculator to accurately measure power during a transient. It has been demonstrated that the ex-core power detectors will provide an adequate power measurement input to the Thermal Margin/Low Pressure (TM/LP) trip for the full spectrum of possible power excursions associated with the CEA withdrawal events with a slight increase in margin to the TM/LP trip setpoint. Thus, the margin of safety is not reduced.

The staff has reviewed the licensee's no significant hazards consideration determination analysis. Based upon this review, the staff believes that the licensee has met the three standards. Based upon the above discussion, the staff proposes to determine that the proposed change does not involve a significant hazards consideration.

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

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

NRC Project Director: Herbert N. Berkow

Illinois Power Company, Soyland Power Cooperative, Inc., Western Illinois Power Cooperative, Inc. (the licensees), Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Date of amendment request: September 23, 1988

Description of amendment request: This proposed amendment would revise Technical Specification Sections 3.8.1.1 and 3.8.1.2, which are the Limiting Conditions for Operation specified for the AC electrical power sources, to change the number of gallons of fuel oil specified for the Division II diesel generator (1B). These Technical Specifications indicate the minimum amount of diesel fuel that should be available for the diesel generators. The licensees have requested to change the number of gallons of fuel oil specified for the Division II diesel generator (1B) from 41,500 to 45,000.

The licensees have prepared a plant modification to replace the Fuel Pool Cooling and Cleanup (FC) System pump motors (1A and 1B) and remove the associated LOCA shunt trips. This modification is in accordance with their commitment "Until the first refueling, the pump motors will be tripped on a LOCA signal.... By the first refueling, replacement motors qualified to the maximum environment conditions will be installed and the LOCA-trip signal will be removed." Removing the associated LOCA shunt trips and ensuring the FC pump motors are qualified to operate in a post-LOCA environment allows the pump motors to be regarded as safety-related essential loads powered from the Class 1E emergency busses. This effectively increases the maximum expected emergency loading for the associated diesel generators (1A and 1B). The resultant increase in the maximum expected loading thus requires a revision of the minimum fuel oil volume specified in the Technical Specifications to ensure that the diesels are capable of supplying and maintaining emergency power for all essential loads.

Basis for proposed no significant hazards consideration determination: The staff has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to 10 CFR

50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the 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 change does not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed change will ensure that an adequate volume of diesel fuel is available for the diesel generator (1B) to perform its intended function in mitigating the consequences of the design basis accident while carrying the maximum expected load (including the associated FC pump motor). The increased maximum expected loading for the diesel generator(s), resulting from the plant modification, does not exceed the rated capacity of the diesel generators.

The impact of the proposed change is confined to two areas of concern: diesel generator operability and the ability to maintain an adequate supply of high quality cooling water in the spent fuel storage pool(s) under post-accident conditions. The changes associated with the plant modification have been evaluated and found to have no adverse impact on the diesel generators' capability to perform their intended function during or following a design basis accident (DBA-LOCA). With respect to any concerns regarding the spent fuel storage pool, including the FC pump motors as essential loads, will ensure that an FC pump is available for cooling and maintaining the volume and quality of water in the spent fuel storage pools under post-accident conditions. The proposed change therefore does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve a significant reduction in a margin of safety because the increased minimum amount of diesel fuel to be stored for diesel generator 1B is well within the storage capacity of the fuel storage tank. In addition, the added electrical load requiring the extra amount of diesel fuel does not cause the maximum expected load for diesel generator 1B to exceed its rated capacity. The electrical loading and fuel storage demand for diesel generator 1B will still be in compliance with the original design requirements.

For the reasons stated above, the staff believes this proposed amendment involves no significant hazards consideration.

Local Public Document Room location: Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727

Attorney for licensees: Sheldon Zabel, Esq., Schiff, Hardin and Waite, 7200 Sears Tower, 233 Wacker Drive, Chicago, Illinois 60606

NRC Project Director: Daniel R. Muller

Long Island Lighting Company, Docket No. 50-322, Shoreham Nuclear Power Station, Suffolk County, New York

Date of amendment request: May 19, 1988

Description of amendment request: This amendment would revise Technical Specifications 3.5.2a.2.b, Emergency Core Cooling Systems - Shutdown, and 3.5.3.1b.3 Suppression Pool Water Level, to read "...equivalent to an indicated level of 11.5 feet" rather than "...equivalent to a level of 9 feet". This change reduces the potential for operator misinterpretation but does not affect the minimum 100,000 gallons of water availability requirement for the condensate storage tank.

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 10 CFR 50.92 the licensee has reviewed the proposed changes and has concluded as follows:

The proposed change does not involve a significant hazards consideration because operation of Shoreham Nuclear Power Station - Unit 1 in accordance with this change would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. This change merely clarifies and identifies the Condensate Storage Tank (CST) level (indicated level) which meets or exceeds the technical specification requirement of maintaining 100,000 gallons of water available for Core Spray system use. The CST low-low level alarm (since May 19, 1982) has always been set at an indicated level of 13 feet as measured from the bottom of the tank. This was always the intended level which was to be used for ECCS reserve.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. It has been determined that a new or different kind of accident will not be possible due to this change. Design documentation specifically calls out a low-low level alarm and a CST transfer pump trip at an indicated level of 13 feet of tank evaluation. Without the foregoing pumps to drain the tank, the ECCS systems are the only users of the water volume below the 13 foot level. If the suction line elevation (approximately 1.75 ft.) is deducted from the 13 ft., a useable volume of 11.25 ft. is achieved. This is equivalent to an approximate available volume of 133,800 gallons.

(3) involve a significant reduction in a margin of safety. The use of an 11.5 ft. indicated level as proposed in the technical specification change clarifies the 1.5 foot (i.e., 13 ft.-11.5 ft) operational deviation that has always existed. If the CST transfer pumps do not deenergize - due to malfunction - at the 13 ft. level, the operator is permitted the same period of time to deenergize the pumps and not place himself in a technical specification violation.

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

Local Public Document Room location: Shoreham-Wading River Public Library, Route 25A, Shoreham, New York 11786-9697

Attorney for licensee: W. Taylor Reveley, III, Esq., Hunton and Williams, P. O. Box 1535, Richmond, Virginia 23212

NRC Project Director: Walter R. Butler

Long Island Lighting Company, Docket No. 50-322, Shoreham Nuclear Power Station, Suffolk County, New York

Date of amendment request: June 13, 1988

Description of amendment request: The amendment would delete Figure 6.2.1-1, "Corporate-Nuclear Organization," and Figure 6.2.2-1, "Unit Organization," from the Technical Specifications and revise sections 6.2.1 and 6.2.2 to include appropriate changes to the administrative control requirements.

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 considerations if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability of 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 Long Island Lighting Company (LILCO) reviewed the proposed change and determined, and the NRC staff agrees, that:

(1) The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated because deletion of the organization charts from the Technical Specifications does not affect plant operation. The NRC will continue to be informed of organizational changes through other required controls.

• 10 CFR 50.34(b)(6)(i) requires that the applicant's organizational structure be included in the Final Safety Analysis Report. Chapter 13 of the SNPS Final Safety Analysis Report provides a description of the LILCO/SNPS organization and detailed organization charts.

• As required by 10 CFR 50.71(e), LILCO submits annual updates to the FSAR.

• Appendix B to 10 CFR 50 and 10 CFR 50.54(a)(3) govern changes to organization described in LILCO's Quality Assurance Program.

LILCO is mindful that some organizational changes may require prior NRC approval. Also, it is LILCO's practice to inform the NRC of organizational changes affecting the nuclear facility prior to implementation. LILCO intends to continue this practice for future organizational changes.

(2) The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed change is administrative in nature, and no physical alterations of plant configuration or changes to setpoints or operating parameters are proposed.

(3) The proposed amendment does not involve a significant reduction in a margin of safety. Through the Company's strong Nuclear Quality Assurance Program and its commitment to maintain only qualified personnel in positions of responsibility, it is assured that safety functions performed by the nuclear organizations will continue to be performed at a high level of performance.

Accordingly, the Commission proposes to determine that the proposed license amendment does not involve a significant hazards consideration.

Local Public Document Room location: Shoreham-Wading River Public Library, Route 25A, Shoreham, New York 11786

Attorney for licensee: W. Taylor Reveley, III, Esq., Hunton and Williams, P. O. Box 1535, Richmond, Virginia 23212

NRC Project Director: Walter R. Butler

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

Date of amendment request: September 21, 1988

Description of amendment request: The proposed amendment would change the Technical Specifications by deleting the Ammonia Detection System. Ammonia detection would be provided by the Broad Range Toxic Gas Detection System which is currently in the Technical Specifications.

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 Broad Range Toxic Gas Detection (BRTGD) system provides toxic gas chemical protection for the control room. The BRTGD system detects ammonia and thereby duplicates the function of the ammonia detection system. Deleting the ammonia detection system will not significantly reduce the protection to the control room envelopment from an ammonia toxic chemical release. The BRTGD system will isolate the control room before the Immediately Dangerous to Life and Health (IDLH) concentrations for ammonia is reached. The technical specifications for the BRTGD system is equivalent to those for the Ammonia Detection System and either system provides for control room isolation as the availability of the system declines. Deletion of the ammonia system as a duplicate to the BRTGD system does not involve a significant increase in the probabilities or consequences of any accident previously evaluated.

The function of both the BRTGD and ammonia systems is solely to isolate the control room in the unlikely event of a toxic chemical release in the area. The BRTGD and ammonia system do not provide any other protective function. Since the BRTGD system will provide ammonia detection, the deletion of the ammonia detection system will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The ammonia detection system setpoint is 50 ppm ammonia and will respond in .33 seconds for concentration at and above 50 ppm. The BRTGD system responds at different times for any concentration above environmental background; the higher the concentration, the faster the response time. For concentration of ammonia of concern at the control room, the BRTGD system will have a faster response time than the ammonia system and while the BRTGD will respond below 50 ppm, the ammonia detection system will not. Such a comparison evaluation demonstrates equivalent or better protection by the BRTGD system. Therefore, the deletion of the ammonia detection system will not involve a significant reduction in a margin of safety.

Based on the above, the staff proposes to determine that the amendment does not involve a no 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

Dates of amendment request: October 19, 1988, as supplemented October 31, 1988

Description of amendment request: The amendment would authorize the sale and leaseback of an individual interest in the Grand Gulf Nuclear Station, Unit 1 (GGNS Unit 1).

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.

a. The proposed change will not increase the probability or consequences of an accident previously evaluated. As a result of the proposed amendment, there will not be physical changes to the facility, and all Operating Procedures, Limiting Conditions for Operation, Limiting Safety System Settings, and Safety Limits specified in the Technical Specifications will remain unchanged. SERI will continue in its present role under the Operating Agreement and Ownership Agreement. There will be no changes to the operating organization or personnel as a result of the transaction(s) described herein.

b. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The design and design bases of GGNS Unit 1 will remain the same. Therefore, the current plant safety analyses will remain complete and accurate in addressing the licensing basis events and in analyzing plant response and consequences. Further, the Operating Procedures, Limiting Conditions for Operation, Limiting Safety System Settings, and Safety Limits specified in the Technical Specifications are not affected. As such, the plant conditions for which the design basis accident analyses were performed are still valid.

c. The proposed amendment will not involve a reduction in any margin of safety. Plant safety margins are established through Limiting Conditions for Operation, Limiting Safety System Settings, and Safety Limits specified in the Technical Specifications. Because there will be no change to either the physical design of the plant or to any of these settings and limits, there will be no change to any of the margins of safety.

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

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

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

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

NRC Project Director: Elinor G. Adensam

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: October 5, 1988

Description of amendment request: The proposed amendment would change Technical Specification (TS) 3/4.1.1.3, "Moderator Temperature Coefficient," to allow a more negative moderator temperature coefficient in the Limiting Condition for Operation, TS 3.1.1.3b, and in the associated Surveillance Requirement, TS 4.1.1.3b.

Basis for proposed no significant hazards consideration determination: The Moderator Temperature Coefficient (MTC) represents the relationship between the reactor coolant system (RCS) temperature and core reactivity. Near the end of the operating cycle the MTC is strongly negative, that is, decreasing RCS temperature causes a substantial increase in core reactivity. Thus, for accidents that involve a significant decrease in RCS temperature, such as a steam line break accident, the MTC strongly influences the severity of the accident. The purpose of TS 3/4.1.1.3b is to assure that the facility will not operate with an MTC more negative than the value incorporated in the safety analyses. The following MTC values are presently in the TS:

- TS 3.1.1.3b - The Limiting Condition for Operation (LCO) for the end-of-life MTC is -4.0×10^{-4} delta K/K/° F. Should the MTC be more negative than the LCO MTC, the reactor would have to be shutdown within 12 hours.
- TS 4.1.1.3b - The Surveillance Requirement (SR) MTC for the end-of-life MTC is -3.1×10^{-4} delta K/K/° F. The SR MTC must be measured within 7 effective full power days (EFPD) after reaching an equilibrium RCS boron concentration of 300 ppm. If the SR MTC is more negative than -3.1×10^{-4} delta K/K/° F, the MTC must be remeasured at least every 14 EFPD during the remainder of the fuel cycle.

Accident analyses do not explicitly input an MTC, but rather a constant moderator density coefficient (MDC). Converting the MDC used in the accident analyses to an MTC is a simple calculation which accounts for the rate of change of moderator density with temperature at the conditions of interest; namely, hot full power. In addition, the MTC that is measured must be corrected to reflect the assumptions used in the safety analysis which includes control rod positions. In this regard, the MDC used in the Millstone Unit 3 accident analysis would be equivalent to an MTC of -5.5×10^{-4} delta K/K/° F.

Westinghouse has recently developed a refined methodology for comparing the measured MTC with the accident analysis MDC. The method developed by Westinghouse is documented in WCAP-11951, "Safety Evaluation Supporting a More Negative EOL Moderator Temperature Coefficient Technical Specification for the Millstone Nuclear Power Station Unit 3," September 1988. By using the methodology of WCAP-11951 for Millstone Unit 3, the following changes are proposed by the licensee for TS 3/4.1.1.3b:

- TS 3.1.1.3b - The LCO MTC would be changed from -4.0 to -4.75×10^{-4} delta K/K/° F.
- TS 4.1.1.3b - The SR MTC would be changed from -3.1 to -4.0×10^{-4} delta K/K/° F.

No change in the safety analysis is involved and the safety analysis MTC value of -5.5×10^{-4} delta K/K/° F is still considered bounding.

Title 10 CFR Part 50, Section 50.92 contains standards for determining whether a proposed license amendment involves significant hazards considerations. In this regard, the proposed changes to the TS does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change to the LCO and SR MTCs provide adequate assurance that Millstone Unit 3 will not be operated with an MTC more negative than the equivalent MDC assumed in the safety analysis. The proposed license amendment does not create the possibility of a new or different kind of accident since no changes to plant equipment or operating modes are involved. Finally, no safety margins are reduced since there are no changes in the safety analyses.

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

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

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

NRC Project Director: John F. Stolz

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

Date of amendment request: December 2, 1986 as supplemented January 9, 1988 and September 30, 1988.

Brief description of amendment: In accordance with the requirements of 10 CFR 73.55, the licensee submitted an

amendment to the Physical Security Plan for the Fort Calhoun Station, Unit 1, to reflect recent changes to that regulation. The proposed amendment would modify paragraph 3.C of Facility Operating License No. DPR-40 to require compliance with the revised plan.

Basis for proposed no significant hazards consideration determination: On August 4, 1986 (51 FR 27817 and 27822), the Nuclear Regulatory Commission amended Part 73 of its regulations, "Physical Protection of Plants and Materials," to clarify plant security requirements to afford an increased assurance of plant safety. The amended regulations required that each nuclear power reactor licensee submit proposed amendments to its security plan to implement the revised provisions of 10 CFR 73.55. The licensee submitted its revised plan on December 2, 1986, with additional information on January 9, 1988 and September 30, 1988, to satisfy the requirements of the amended regulations. The Commission proposes to amend the license to reference the revised plan.

In the Supplementary Materials accompanying the amended regulations, the Commission indicated that it was amending its regulations "to provide a more safety conscious safeguards system while maintaining the current levels of protection" and that the "Commission believes that the clarification and refinement of requirements as reflected in these amendments is appropriate because they afford an increased assurance of plant safety."

The Commission has provided guidance concerning the application of the criteria for determining whether a significant hazards consideration exists by providing certain examples of actions involving no significant hazards considerations and examples of actions involving significant hazards considerations (51 FR 7750). One of these examples of actions involving no significant hazards considerations is example (vii) "a change to conform a license to changes in the regulations, where the license change results in very minor changes to facility operations clearly in keeping with the regulations." For the foregoing reasons, the Commission proposes to determine that the proposed amendment involves no significant hazards consideration.

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

Attorney for licensee: LeBoeuf, Lamb, Leiby, and MacRae, 1333 New Hampshire Avenue, NW., Washington, DC 20036

NRC Project Director: Jose A. Calvo

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

Date of amendment request: September 29, 1988

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) to: (1) delete the primary containment isolation valves and instrumentation associated with the permanent removal of the RHR head spray piping and (2) modify the reportability requirements for seismic monitor XR-VA-151 whenever the reactor head has been removed.

Basis for proposed no significant hazards consideration determination: Limerick Units 1 and 2 are BWR-4 reactors. All BWR-4s were designed with three penetrations at the top of the reactor vessel head, one four-inch and two six-inch penetrations. One of the six-inch penetrations was intended to provide a water spray to the space at the top of the reactor vessel. Located above the reactor core are the steam separators and dryers. It was postulated that during cooldown of the reactor system, a spray of water would be required to cool the large mass of metal in the separators and dryers. The source of water was primary coolant from the residual heat removal (RHR) system. Over 15 years ago, it was found that this RHR head spray was not needed and is no longer used during system cooldown. Keeping the system in place poses a number of potential safety and economic disadvantages. Each time the reactor vessel head is removed (e.g., during refueling), the array of piping and valves has to be disassembled and removed and then reinstalled after the head is replaced. Since the piping contains "stagnant" primary coolant at system temperature and pressure, there exists the potential for intergranular stress corrosion cracking of the many welds in the system, increasing the potential for leakage. Consequently, these welds are subject to the augmented inspection requirements of NUREG-0313. The piping constitutes one more potential source for a high energy line break and for pipe whip. Since all of the BWR-4s have demonstrated that there is no need for the RHR head spray and since removing the piping inside containment enhances plant safety, the NRC has approved removal of this system in most BWR-4s. Limerick Unit 1 is one of the minority that has so far retained the RHR head spray system.

The proposed application requests NRC approval for removal of the RHR head spray piping and associated valves, and for blanking off the associated primary containment penetration and the existing reactor head spray piping stubout used for the head spray. At present, there is a seismic monitor (XR-VA-151) located on the head spray piping. The licensee proposes to relocate this monitor to place it directly on the reactor head at the nozzle presently used for the head spray piping penetration. This seismic monitor has to be removed (and subsequently replaced) each time the vessel head is removed. The present TSs require that whenever the seismic monitor is inoperable - and disconnecting the monitor renders it inoperable - a special report has to be submitted to the Commission.

The proposed changes to the TSs would eliminate all references to the RHR head spray piping and isolation valves, instruments and controls. The changes would also eliminate the requirement for a special report to the Commission when the seismic monitor XR-VA-151 is inoperable because it had to be disconnected to remove the reactor vessel head.

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 consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The licensee's analysis contained in their September 29, 1988 letter states the following in response to the three NRC criteria referenced above with respect to the changes to the TSs to delete the isolation valves and instrumentation associated with permanent removal of the RHR head spray piping:

(1) Operation of the plant under the proposed Technical Specifications after removal of the RHR Head Spray piping and associated valves along with blanking the associated primary containment penetration, would not involve a significant increase in the probability or consequences of an accident previously evaluated.

Three areas were previously evaluated in the FSAR regarding the reactor head spray piping:

- Primary Containment Isolation - FSAR Section 6.2
- Seismic Analysis - FSAR Section 3.7.4
- Pipe Whip Analysis - FSAR Section 3.6

The Reactor Head Spray piping removal modification was reviewed and found to be acceptable in the above referenced FSAR areas.

• The primary containment isolation will be maintained after removal of the Reactor Head Spray piping by welding a closure on the outboard containment side of penetration 17. The penetration is included in the inservice inspection (ISI) program and the integrity of the welded closure will be verified by periodic testing.

• Seismic Category I piping, hangers and snubbers on the RHR Head Spray would be removed by the proposed modification. Stress calculations have been reviewed and appropriately revised to assure that any remaining components are not impacted.

• Any potential pipe whip problems would be eliminated by removal of the pipe and pipe supports, as proposed. Further, Licensee has reviewed the potential effects of the proposed removal in previous evaluations in the areas of Fire Protection, Electrical Separation, Environmental Qualification, Inservice Inspection, and Piping Stresses. Evaluations in these areas did not uncover any areas of safety significance.

Based on these reviews, the Licensee concludes that the RHR Head Spray modifications do not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the plant under the proposed Technical Specifications after removal of the Reactor Head Spray System along with blanking the primary containment penetration would not create the possibility of a new or different kind of accident from any accident previously evaluated.

Removal of the RHR Head Spray piping and blanking the primary containment penetration eliminates the piping from being a potential pipe whip problem and removes the containment penetration as a potential leakage source. No credit has been taken for the RHR Head Spray in the mitigation or prevention of an accident, therefore, the modification does not create the possibility of new or different kind of accident from any accident previously evaluated.

(3) Operation of the plant under the proposed Technical Specifications after removal of the RHR Head Spray and associated valves along with blanking the primary containment penetration, would not involve a significant reduction in a margin of safety.

The integrity of the reactor pressure boundary after removal of the RHR Head Spray would be maintained by a blank flange installed over the Reactor Head Spray nozzle. The reactor pressure boundary would then become part of the Inservice Inspection (ISI) program and would be hydrostatically tested each time the reactor head is reinstalled on the reactor vessel. The primary containment penetration will be welded closed on the outboard side of the containment penetration and will be periodically tested for integrity during scheduled integrated leak rate testing. The seismic category I piping, hangers and snubbers and containment isolation valves on the Head Spray piping would be removed by the proposed modification.

Therefore, the RHR Head Spray modification would not involve a reduction in a margin of safety. Based on the three standards discussed above, operation of the facility subsequent to removal of the RHR Head Spray along with the associated primary containment isolation valves, involves No Significant Hazard Considerations.

The licensee separately evaluated the deletion from the TSs of the special report when seismic monitor XR-VA-151 is inoperable solely because it has to be disconnected to remove the reactor vessel head. With respect to the three NRC criteria in 10 CFR 50.92, the licensee stated:

(1) Operation of the plant under the proposed Technical Specifications in regard to changing the operability reporting requirements for seismic monitor XR-VA-151 whenever the reactor head has been removed, would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Seismic monitor would continue to function under the proposed amendment whenever the reactor head is installed on the reactor vessel. When the reactor head has been removed from the reactor vessel, the seismic monitor will become inoperable by necessity and presently is reportable after 30 days under the existing Technical Specifications.

The purpose of the existing requirement is to report unexpected seismic monitor malfunctions during periods when monitors are required to be operable. Eliminating the requirement for submission of a special report when only one seismic monitor becomes inoperable for more than 30 days, during the course of normal activities taking place with the reactor head removal, will not affect the reporting requirements for the monitor under any other operating conditions. Following the reinstallation of the reactor head, the seismic monitor will be reconnected and its operability re-established. The reporting requirements for other seismic monitors would not be affected by this proposed change.

The intent of the specification for reporting seismic monitor malfunction would continue under the proposed amendment. Lack of a report whenever the reactor head is removed, does not affect the intent of the specification which is to accrue data on unexpected seismic monitor malfunctions and on the reliability of the monitors, rather than on intentional disconnections of a monitor.

Therefore, deletion of the reportability requirement under these expected conditions would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the plant under the proposed Technical Specifications in regard to changing the operability reporting requirements for Seismic Monitor XR-VA-151 whenever the reactor head has been removed would not create the possibility of a new or different kind of an accident from any accident previously evaluated.

After completion of the proposed Head Spray removal modification, a blanking flange on the reactor nozzle would maintain the reactor pressure boundary. Seismic monitor XR-VA-151 would be remounted at the time to the new blanking flange on top of the reactor vessel head. No other changes are being proposed for seismic monitor XR-VA-151. Also, the reportability requirements would not be changed for any other monitor except XR-VA-151 under the proposed amendment. Changing the reportability requirements for seismic monitor XR-VA-151 when the reactor head has been removed does not affect any plant safe shutdown capabilities.

Therefore, elimination of the reportability requirements without making changes to the location or to the normal operability status of the seismic monitor would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the plant under the proposed Technical Specification in regard to changing the operability reporting requirements for Seismic Monitor XR-VA-151 whenever the reactor head has been removed would not involve a significant reduction in a margin of safety.

The Seismic Monitoring System provides information to the operators after a seismic event and does not perform any direct plant shutdown function or affect plant operation. When monitor XR-VA-151 becomes inoperable during times the RPV head is removed, it does not provide any information following a seismic event occurring during that period. Other monitors in the plant remain operable and would provide this information. The lack of post-seismic data from seismic monitor XR-VA-151 would remain the same whether or not a special report was submitted to the Commission. Elimination of the Special Report when seismic monitor XR-VA-151 is inoperable during times when the RPV head is removed, does not involve a significant reduction in a margin of safety. The lack of seismic information from seismic monitor XR-VA-151 after any seismic event when the reactor head is removed, would not affect the safety of the plant. Seismic monitors provide information to reinforce and verify previous seismic calculations. Other monitors in the plant would provide this information when XR-VA-151 is not operable.

Based on the three standards discussed above, operation of the facility after changing the seismic monitor reportability requirements in the Technical Specifications, involves No Significant Hazards Considerations.

The staff has reviewed the licensee's analyses and agrees with it. Therefore, we conclude that the amendment satisfies the three criteria listed in 10 CFR 50.92(c). Based on that conclusion, the staff proposes to determine that the proposed license amendment does not involve a significant hazards consideration.

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

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

NRC Project Director: Walter R. Butler

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

Date of amendment request: April 14, 1988

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) Section 4.6.G.1 to resolve a conflict with the corresponding Bases section. Section 4.6.G.1 specifies surveillance requirements to verify jet pump performance. The revision would reduce the maximum permissible recirculation loop flow imbalance between recirculation loops from 15 percent to 10 percent when the recirculation pumps are operated at the same speed, which would then agree with the limits stated in the TS Bases section. The 10 percent value is consistent with the TS Bases, NRC staff and industry guidance.

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 previously evaluated; or (3) Involve a significant reduction in a margin of safety.

The licensee has determined, that the proposed TS change will not involve a significant hazards consideration. The proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated in that the effect is to bring the TS section into agreement with the actual practice and the TS Bases section. The proposed change is administrative in nature and as such does not involve hardware or procedural changes to the facility. The proposed change will not create the possibility of a new or different kind of accident since it does not involve an actual change to present operating criteria and as stated previously does not involve any facility hardware or procedural changes. The proposed change does not involve a reduction in the margin of safety because the change is administrative in nature. In fact, the proposed change increases the probability that a jet pump failure will be promptly identified by the operators since the effect is to reduce the jet pump performance surveillance acceptance limit. The 10 percent figure complies with the General Electric Company's Service Information Letter No. 330, which verifies that the 10 percent value is the proper limit.

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

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

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. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego, New York

Date of amendment request: September 13, 1988

Description of amendment request: The licensee provided, in part, the following description: The application for amendment proposes changes to page 4, Table 3.1-1 and Table 3.2-1 of the Technical Specifications (TS). The change to page 4 would delete the reactor protection scram bypass from the definition of Startup/Hot Standby. The change to Table 3.1-1 would delete the requirement for a reactor scram on main steam isolation valve (MSIV) closure in the refuel and the startup modes. Also note 3 to this Table, which established 1005 psig as the reactor pressure below which the scram is bypassed, will be deleted. The change to Table 3.2-1 involves adding a reference to note 7 for the low condenser vacuum trip of the MSIVs to indicate that the trip is functional only in the run mode. Note 8 to Table 3.2-1, which already refers to the low condenser vaccum trip of the MSIVs, would be changed to read, "Bypassed when mode switch is not in run mode and turbine stop valves are closed." This would remove the requirement that reactor pressure be less than 1005 psig before the bypass occurs.

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 evaluated the proposed amendment against the standards in 10 CFR 50.92 and the licensees' findings are summarized below:

1. The proposed change does not increase the probability or consequences of an accident previously evaluated. Pressure switches set at 1005 psig were installed a few years ago after instability was observed in an early European Boiling Water Reactor during its startup. However, a series of recent reactivity and pressue perturbation tests conducted as part of the startup test program at Browns Ferry (a typical BWR 4 design of the FitzPatrick type) showed that following the initial disturbance, all parameters returned to steady state values and the reactor stabilized.

In addition, since the switches are set to bypass up to the normal reactor operating pressure of 1005 psig, the pressures which would allow the scram on MSIV closure and main steam line isolation on low condenser vacuum when the turbine stop valves are closed are outside the range of pressures for the refuel and startup modes. Thus, scram and isolation functions are bypassed and the pressure switches are not necessary. The consequences of inadvertent MSIV closure in the refuel or startup modes at or below 1005 psig will remain unchanged with the removal of the switches. In the startup mode, the reactor power is between approximately 0-15% of full power and the peak reactor pressure and the critical power ratio responses are significantly below the limits established for transients during full power operation.

In startup mode, the Intermediate Range Monitor (IRM) subsystem and the Average Power Range Monitor (APRM) subsystem provide signals to the Reactor Protection System (RPS) to shutdown the reactor. If MSIV closure occurs while the reactor is in the startup mode, the reactor will scram on high neutron flux or high reactor pressure. The overpressure protection analysis, for the limiting event of MSIV closure at

100% power terminated by the high neutron flux scram, provides the bounding analysis for the pressure transient. If a loss of condenser vacuum event occurs during refuel or startup modes, the turbine bypass valves would close to isolate the condenser, and operator action can be taken to manually close the MSIVs if necessary. Therefore, removal of pressure switches and deletion of scram and isolation functions does not increase the probability or the consequences of an accident.

2. The proposed change will not create the possibility of a new or different kind of accident from any previously evaluated. The purpose for which the pressure switches were installed does not exist (as discussed above). With the switches set for bypass at 1005 psig (which is above the full range of reactor pressures for refuel and startup modes), scram on MSIV closure and isolation on low condenser vacuum during refuel and startup modes of operation are bypassed. Therefore, the pressure switches are not used for any safety function and no new or different kind of accident can be created by the removal of these switches and deletion of the scram and isolation functions.

3. The proposed amendment will not involve a significant reduction in the margin of safety. The current setpoint for the pressure switches bypass the scram and isolation functions for the full range of reactor pressures in the refuel and startup modes. Furthermore, the operating limits of the plant are not determined by the setpoint of these switches. The limiting plant transients are still those initiated from full power operation and not from operation in the refuel or startup modes with the scram and isolation bypass. Therefore, the operating limits and the limiting safety system settings remain unchanged and the margin of safety is not reduced.

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

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

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: August 16, 1988

Description of amendment request: The proposed amendment would revise the Technical Specifications to streamline the Monthly Operating Report to conform to that of the Standard Technical Specifications. Redundancy within the current Monthly Operating Report will be eliminated, and the reporting of safety and relief valve challenges will be provided on a more frequent basis by including this information within the monthly rather than the annual report. The Monthly Operating Report will continue to provide the information outlined in Regulatory Guide 1.16.

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 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 an accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

The licensee made the following analysis of these changes:

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

Response

The proposed changes described and evaluated above do not involve a significant increase in the probability or consequences of an accident previously evaluated. Revising the wording of the section on Monthly Operating Reports does not alter any system or subsystem and will not change the conclusions of either the FSAR or SER accident analysis.

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

Response

These changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve any changes to the hardware, operability, surveillance, or record-keeping requirements of the facility. In addition, safety and relief valve challenges will subsequently be reported on a more frequent basis.

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

Response

The proposed changes do not involve a reduction in a margin of safety since they do not in any way affect the availability, operability, or surveillance requirements of any equipment within the facility. The changes revise the wording of the IP3 Technical Specifications section concerning monthly reporting requirements to conform to that of the Standard Technical Specifications.

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: August 23, 1988

Description of amendment request: By NRC Generic Letter 84-13 dated May 3, 1984, "Technical Specifications for Snubbers," an option was provided to delete snubber listings from a plant's Technical Specifications. The proposed amendment will revise the Technical Specifications by deleting the snubber listing (Table 3.13-1) while maintaining operability, surveillance, and record-keeping requirements.

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 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 made the following analysis of these changes:

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

Response

The proposed changes described and evaluated above do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes do not alter any system or subsystem and will not change the

conclusions of either the FSAR or the SER accident analysis.

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

Response

These changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes do not involve hardware or procedural changes to the facility. Deletion of the snubber listing from the Technical Specifications does not affect safety-related snubber operability, surveillance or record-keeping requirements, and thus cannot create the possibility of a new or different kind of accident. A listing of all the safety-related snubbers is maintained as part of the surveillance performance test procedures, which is a controlled document.

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

Response

The proposed changes do not involve a reduction in a margin of safety since they do not in any way reduce the availability of the snubbers that are provided to ensure that the structural integrity of the reactor coolant and all other safety-related systems are maintained during and following a seismic or other event that induces dynamic loads.

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

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

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

NRC Project Director: Robert A. Capra, Director

Public Service Company of Colorado, Docket No. 50-267, Fort St. Vrain Nuclear Generating Station, Weld County, Colorado

Date of amendment request: September 23, 1988

Description of amendment request: This amendment request results from the licensee's need to have a 500 curie source of cesium-137 on site to perform calibration of a high range detection instrumentation. Amendment 41 to the Fort St. Vrain Technical Specifications directed Public Service Company of Colorado to, at a future time, replace the listing of specific isotopes with a statement similar to that now requested in 2.c.(4). The current Radiological Control Program maintains adequate control of the use and storage of calibration sources. This will serve to place the Fort St. Vrain License in a format more similar to the recently issued Licenses.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The licensee provided an analysis that addressed the above three standards in the amendment application as follows:

The proposed change does not involve a significant hazards consideration because operation of FSV in accordance with this change would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The use of sources for calibration purposes does not effect the design or function of any plant system/component. The updating of this license condition to not specify individual isotopes will allow more flexibility, and is being done based on an NRC recommendation.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The calibration of detection instrumentation does not create the possibility of any accident different from those already analyzed. Non-specific designation of the calibration sources will not create any new failure modes.

(3) involve a significant reduction in a margin of safety. There is no margin of safety associated with calibration source strength.

Further, reformatting the license does not alter the requirements expressed in the License.

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

Local Public Document Room location: Greeley Public Library, City Complex Building, Greeley, Colorado

Attorney for licensee: James K. Tarpey, Public Service Company Building, Room 900, 550 15th Street, Denver, Colorado 80202

NRC Project Director: Jose A. Calvo

Public Service Company of Colorado, Docket No. 50-267, Fort St. Vrain Nuclear Generating Station, Weld County, Colorado

Date of amendment request: October 13, 1988

Description of amendment request: The proposed Amendment would modify Technical Specification Section LCO 4.4.1, which provides a listing of the Plant Protective System (PPS) instrumentation parameters and the associated bases. The PPS is the reactor protective circuitry and the circuitry oriented towards protecting various plant components from major damage.

The Technical Specification LCO 4.4.1 has been modified to clarify the time permitted to reset trip setpoints per the detector decalibration curve, Figure 3.3-1, for the linear channel - high neutron flux channels following a power reduction.

If the linear channel - high neutron flux channels are outside their Allowable Values, they must be considered inoperable and the appropriate actions apply. The linear channel - high neutron flux RWP and scram will be available but may not be set properly. The Technical Specifications currently requires a plant shutdown within 12 hours. There are various plant situations where power level is automatically reduced and the applicable Trip Setpoints for the linear channel - high neutron flux channels change.

To avoid unnecessary shutdown requirements after control rod runback or power reduction events, the licensee proposes that an action be added to the Technical Specifications that allows 12 hours after a power reduction to regain compliance with Figure 3.3-1 for linear channel - high neutron flux. This added action provides a reasonable period of time to regain compliance, either by adjusting the Trip Setpoints or by changing power level. During this time, the linear channel - high neutron flux RWP and scram, (which may improperly set), and the reheat steam temperature-high scram provide protection against an unexpected increase in power level. The likelihood of a rod withdrawal accident (for which these scram parameters provide protection) is small. The 24 hour orderly shutdown requirement reduces rapid transients on plant components and is consistent with actions included in the Technical Specification Upgrade Program.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of

a new or different kind of accident from any accident previously evaluated: or (3) involve a significant reduction in a margin of safety. The licensee provided an analysis that addressed the above three standards in the amendment application as follows:

The proposed amendment does not involve a significant hazards consideration because operation of the Fort St. Vrain Nuclear Generating Station in accordance with this change would not:

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

The linear channel - high neutron flux parameters are part of the Plant Protective System (PPS). The primary function of the linear channel - high neutron flux parameters is to provide a scram prior to reactor power exceeding 140% of rated power. Additional protection is provided by a rod withdrawal prohibit prior to reactor power exceeding 120% of rated power. These high neutron flux scram and RWP actions are backed up by the PPS reheat steam temperature - high scram. Section 14.2.2 of the FSAR analyzes accident scenarios that would produce reactor power levels of 140% of rated power. The condition that is most likely to cause an increase in power level of this nature is a rod withdrawal accident. Section 14.2.2.6 analyzes maximum worth control rod pair withdrawal at full power. Included are scenarios where the reactor is scrammed 88 seconds and 105 seconds after accident initiation by the reheat steam temperature - high scram. These accident analyses are not modified by this amendment.

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

FSAR Section 14.2.2 contains the analysis of core reactivity accidents. Permitting a reasonable amount of time to regain compliance with Figure 3.3-1 for the linear channel - high neutron flux channels and a reasonable amount of time to shut down the reactor in an orderly manner does not change that analysis.

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

The margin of safety against an increase in reactivity accident is provided by five protective actions identified in FSAR Section 14.2.2.1. This amendment clarifies the time that is available to regain compliance with Figure 3.3-1 for two of these protective actions following a power reduction that changes the applicable trip setpionts for the linear channels. Any reduction of safety during this time is not significant in that all five protective actions are available. (The RWP and scram for the linear channels may be improperly set on an interim bases.) The effectiveness of the other three protective actions is analyzed in FSAR Section 14.2.2.6. The other protective actions include reheat steam temperature.- high scram. manual scram. and manual actuation of the reserve shutdown system.

In this requested revision to LCO 4.4.1 for the power reduction situation. 12 hours would be permitted to ensure proper trip setpoints for the linear channel - high neutron flux channels. This could include either adjusting the trip setpoints for the lower power level. or increasing reactor power. if appropriate. Also. 24 hours would be permitted to effect an orderly shutdown of the reactor in the unlikely event that compliance with Figure 3.3-1 could not be regained. Interim Technical Specification LCO 3.1.5 permits 4 hours to restore the control rods to an acceptable configuration following a control rod runback. The resetting of the trip setpoints must be done after the control rods are restored to an acceptable configuration. The twelve hours includes time to position the control rods to conform to the requirements of interim Technical Specification LCO 3.1.5.

PSC considers this change to LCO 4.4.1 justified because adequate protective actions remain in place and a rod withdrawal accident is a low probability event. During the interval in which the high neutron flux scram trip setpoint may not be in compliance with Figure 3.3-1. the reheat steam temperature - high scram would be available to protect against an unexpected increase in reactor power. The RWP and scram due to high neutron flux would be available but mat not actuate by the 120% or 140% analyzed values. The manual scram is also available in addition to the automatic scram and RWP actions.

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

Local Public Document Room location: Greeley Public Library. City Complex Building. Greeley. Colorado

Attorney for licensee: J. K. Tarpey. Public Service Company Building. Room 900. 550 15th Street. Denver. Colorado 80202

NRC Project Director: Jose A. Calvo

Public Service Company of Colorado. Docket No. 50-267, Fort St. Vrain Nuclear Generating Station. Weld County, Colorado

Date of amendment request: October 14. 1988

Description of amendment request: The amendment would make certain changes to Section 7 of the Technical Specifications. concerning Administrative Controls. The changes include deletion of the organizational charts from the Technical Specifications as per Generic Letter 88-06 dated March 22. 1988. AC 7.1.1 is reformatted to better conform with the Standard Technical Specifications (based on Westinghouse plants). AC 7.1.1. 7.1.2. 7.1.3. and 7.2 have changes made to position titles reflecting a recent reorganization at Fort St. Vrain.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated: or (2) create the possibility of a new or different kind of accident from any accident previously evaluated: or (3) involve a significant reduction in a margin of safety. The licensee provided an analysis that addressed the above three standards in the amendment application as follows:

A. Specification AC 7.1.1 is revised as follows:

Section 1. "RESPONSIBILITY." was added delegating control room command responsibility and corporate responsibility for overall nuclear plant safety. This section directly correlates to Section 6.1 of the Westinghouse Standard Technical Specifications (STS). NUREG-0452.

Section 2.a. "Onsite and Offsite Organization." was added to provide a directive for the establishment and definition of lines of authority. responsibility. and communication for onsite and offsite organizations. This section directly correlates to Section 6.2.1 of the STS. as amended by Generic Letter 88-06 for deletion of the organization charts from the Technical Specifications (Tech. Specs.).

Section 2.b(1) designates on-duty shift minimum shift crew composition per Table 7.1-1. This section follows the guidelines of Generic Letter 88-06. and incorporates the requirements of Generic Letter 88-06. and incorporates the requirements of current Section 2.a.

Section 2.b(2.3.4) discuss licensed operator on-shift requirements. These sections are added to conform to the formatting effort of this amendment: and include part of the current clarification text following Table 7.1-1 and the requirements of current Section 2.b.

Sections 2.b(5.6.7.8.9) are reformatting and editorial corrections. which incorporate the requirements of current Sections 2.c. 2.d. 2.e. 2.i. and the position titles of the Fort St. Vrain (FSV) reorganization.

Section 2.b(10) was added to delineate those requiring a Senior Reactor Operator's (SRO) license and those requiring a Reactor Operator's (RO) license: and follows the guidelines of Generic Letter 88-06.

Section 2.b(11) discusses shift crew composition and incorporates the requirements of the current final clarification paragraph following Table 7.1-1. which were not included in proposed Sections 2.b(2.3.4).

Section 2.b(12) was added to delineate control room command responsibility in the absence of the Shift Supervisor. and further expounds on proposed Section 1.b.

Table 7.1-1 was relocated within Specification AC 7.1.1. The table retains the same requirements as the current Table 7.1-1.

However, the page is reformatted to include the applicable notes.

Section 3, "TECHNICAL ADVISORS," is a reformatting addition. This section incorporates the requirements of current Section 1.c and 2.f, with position titles per the FSV reorganization.

Section 4, "UNIT STAFF QUALIFICATIONS," is a reformatting addition and incorporates the requirements of current Sections 2.g and 2.h. "upon commencement of commercial operation" is deleted since this stipulation is not necessary.

Section 5, "TRAINING," reformats and incorporates the requirements of current Section 3. "Compliance with Section 5.5 of ANSI 18.1-1971 shall be achieved no later than 6 months following commencement of commercial operation" is deleted since this stipulation is not necessary.

B. Specification AC 7.1.2 is revised as follows:

Section 1, Plant Operations Review Committee (PORC) Membership, is revised to incorporate position titles of the FSV reorganization. No expertise is deleted from the PORC, and the positions meet the description of ANSI N18.1.

Sections 5.a, 5.3, 6.d and 7 contain a position title change only. Responsibility and expertise remain the same.

Sections 3, 4, 9, and 10 contain position title change only. Responsibility and expertise remain the same.

Section 5.k, relative to PORC review of every unplanned onsite release of radioactive material to the environs, is deleted. This requirement is considered to be adequately covered in Sections 5.a through 5.g. Also, several recent plant Technical Specifications have been found not to include this requirement: River Bend 1, Grand Gulf 1, Nine Mile Point 2, and Palo Verde 1. Deletion of Section 5.k is also consistent with the Tech. Spec. Upgrade Program draft.

C. Specification AC 7.1.3 is revised as follows:

Section 1 contains only formatting changes.

Section 2 contains revisions to the Nuclear Facility Safety Committee (NFSC) Membership relative to position titles per the FSV reorganization. The actual membership and areas of responsibility/ expertise remain the same.

Sections 3, 4, and 9 contain a position title change only. Responsibility and expertise remain the same.

Section 10.a is revised to delete the requirement that each NFSC meeting's minutes be approved within 30 days following each meeting. Section 10.b was added to direct the preliminary approval of the NFSC meeting minutes by the Senior Vice President, Nuclear Operations. It also directs the distribution of the minutes to the NFSC members, and approval of the minutes at the next NFSC meeting. These revisions will ensure that the entire NFSC membership will be given the opportunity to vote on the approval of the minutes of the last NFSC meeting.

D. Pages 7.1-20, 7.1-21, 7.1-22, and 7.1-23 are deleted:

Pages 7.1-20 and 7.1-21 contain Table 7.1-1 and its associated notes and clarification information. All this information is included elsewhere in the proposed amendment.

Pages 7.1-22 and 7.1-23 contain the organization charts (Figures 7.1-1 and 7.1-2). These charts are deleted from the Tech. Specs., based on the recommendation of Generic Letter 88-06.

E. Specification AC 7.2 is revised as follows:

Sections b. and d. contain a position title change only. Responsibility and expertise remain the same.

Except as otherwise noted above, this proposed amendment reformats current Administrative Controls Specification 7.1.1 requirements to better conform to STS formatting; deletes organization charts per the guidelines of Generic Letter 88-06; and retitles certain positions in AC 7.1.1, 7.1.2, 7.1.3, and 7.2 to conform to the FSV reorganization begun May 12, 1988.

Based on the above, this proposed change does not involve a significant hazards consideration because operation of FSV in accordance with this change would not: involve a significant increase in the probability or consequences of an accident previously evaluated; create the possibility of a new or different kind of accident from any accident previously evaluated; involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Greeley Public Library, City Complex Building, Greeley, Colorado

Attorney for licensee: J. K. Tarpey, Public Service Company Building, Room 900, 550 15th Street, Denver, Colorado 80202

NRC Project Director: Jose A. Calvo

Public Service Company of Colorado, Docket No. 50-267, Fort St. Vrain Nuclear Generating Station, Weld County, Colorado

Date of amendment request: October 14, 1988

Description of amendment request: The proposed changes would modify SR 5.0 to add a new inservice inspection criterion, SR 5.2.1, Prestressed Concrete Reactor Vessel (PCRV) and PCRV Penetration Overpressure Surveillance and its basis would be modified. The modifications would require testing of the overpressure protection assembly in accordance with Subsection IGV or IWV of Section XI of the ASME Code.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance for the application of criteria for no significant hazards consideration determination by providing examples of amendments that are considered not likely to involve a significant hazards consideration (51 FR 7751). These examples include "(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 change would require the licensee to test the PCRV overpressure assembly in accordance with the widely accepted ASME code. This constitutes an additional and more stringent restriction that is not currently included in the Technical Specifications and is therefore within the scope of the example cited above.

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

Local Public Document Room location: Greeley Public Library, City Complex Building, Greeley, Colorado

Attorney for licensee: J. K. Tarpey, Public Service Company Building, Room 900, 550 15th Street, Denver, Colorado 80202

NRC Project Director: Jose A. Calvo

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

Date of amendment request: September 28, 1988

Description of amendment request: This amendment would revise Technical Specification Tables 2.2.1-1, 3.3.2-1, and 3.3.2-2 to replace the footnote created with the issuance of Amendment 8 (restrictions associated with the hydrogen injection test) with the necessary requirements associated with the installation of a permanent Hydrogen Water Chemistry (HWC) System. These changes would permit the operation of an HWC system by creating two separate main steam line background radiation levels and associated trip setpoints while restricting operation to power levels greater than 20% of Rated Thermal Power.

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

amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. In accordance with 10 CFR 50.92 the licensee has reviewed the proposed changes and has concluded as follows that they do not involve a significant hazards consideration:

Significant Hazards Analysis Consideration

The proposed changes to the Technical Specifications:

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

The only accident scenario which takes credit for the MSL high radiation scram and isolation setpoint is the Control Rod Drop Accident (CRDA) as described in the Updated Final Safety Analysis Report (UFSAR) Section 15.4.9. Specifically, the Main Steam Isolation Valves (MSIVs) are assumed to receive an automatic closure signal at 0.5 seconds after detection of high radiation in the main steam lines and to be fully closed at 5 seconds from the receipt of the closure signal. The MSL radiation monitors are provided to detect a gross failure of the fuel cladding. When high radiation is detected, a trip is initiated to reduce the continued failure of fuel cladding. At the same time, the MSIVs are closed to limit the release of fission products. The trip setting is high enough above background radiation levels to prevent spurious trips yet low enough to promptly detect gross failures in the fuel cladding.

NEDO-10527, Supplement 1, "General Electric Rod Drop Accident Analysis for Large Boiling Water Reactors" dated July 1972 concluded that the consequences of the CRDA are most severe under Hot Standby conditions. Furthermore, the consequences of the CRDA are increasingly less severe above 10 percent power due to a faster Doppler response and a lower rodworth. Finally and most importantly, this report concluded that above 20 percent power the consequences of the CRDA are minimal. Therefore, the Guidelines (Section 8.2.1 and Table 2-1) indicate that the hydrogen injection system should not be operated below the limiting low power setpoint for the CRDA as discussed in the UFSAR. HCGS UFSAR Section 15.4.9 does not actually specify this low power limit; however, Sections 7.7.1.1.5.4 and 7.7.1.1.5.4.1 do-20% of Rated Thermal Power. This limit is known as the Low Power Setpoint (LPSP) and is contained in Technical Specifications 3/4.1.4.1 (Rod Worth Minimizer) and 3/4.1.4.2.(Rod Sequence Control System).

As a result, the MSL radiation monitor setpoint will only be adjusted upward when the hydrogen water chemistry system is operated. HWC system operation is restricted to power levels greater than 20 percent of Rated Thermal Power. This power level differs from the 22 percent of Rated Thermal Power level contained in Amendment 8 for the hydrogen injection test for two main reasons. First, the hydrogen injection test was only a test, the permanent system is a complete, long-term system with the necessary instrumentation, controls and trips to more accurately control hydrogen injection. Since the HWC system is designed in accordance with the Guidelines and utilizes the experience gained during the hydrogen injection test and from systems installed at other utilities, system operation is closely and accurately controlled and monitored. Second, the Guidelines specify that injection should occur at the LPSP and does not require an additional margin. The hydrogen injection test added an additional 2% power margin simply to assure that the system was not operated below the LPSP. The permanent HWC system will contain sufficient controls to assure operation above the LPSP. Therefore, operating the HWC system at HCGS with such a setpoint (i.e. 20% of Rated Thermal Power) provides adequate assurance that the consequences of a CRDA are negligible when the system is in operation.

Furthermore, in order to assure that the setpoint adjustment process itself does not have any impact on the plant, if a power reduction event occurs so that the reactor power is below 20% of Rated Thermal Power without the required setpoint change, control rod motion will be suspended (except for scrams or other emergency conditions) until the necessary setpoints adjustment is adjusted. This restriction further assures that the possibility of a CRDA occurring while the setpoints are being adjusted is precluded.

Therefore, it can be concluded that the proposed changes to the Technical Specifications do not increase the probability or consequences of an accident previously evaluated.

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

The proposed changes do not affect the design of any safety related systems and as such do not affect the performance of any safety related functions. The proposed changes do permit the operation of the station with a new system, namely a hydrogen water chemistry system. However, this system has been extensively analyzed by EPRI, approved for use by the NRC (reference the Guidelines and the associated NRC SER on them cited in Item III above), and is in operation at a variety of utilities including the Dresden-2, FitzPatrick and Duane Arnold stations. Attachment 4 [of the licensee's submittal] contains a graphical comparison of the operation of HWC systems at these and other utilities which have utilized the services of General Electric in the design and operation of the hydrogen injection test and hydrogen water chemistry system.

The decision to seek a permanent change to the HCGS Technical Specifications is plant specific since a change is necessary only if the increase in the MSL radiation levels does not provide an acceptable margin to the MSL radiation monitor setpoint established without operation of a HWC system. Although the operation of a HWC system introduces hydrogen in the recirculation system, this condition has already been analyzed in UFSAR Sections 6.2.5 (Combustible Gas Control System), 10.4.2 (Main Condenser Evacuation System), and 11.3.2.1 (Offgas System). In addition, the level of hydrogen in the offgas system is controlled and monitored in accordance with Technical Specifications 3/4.3.7.11 and 3/4.11.2.6, respectively.

PSE&G is evaluating the impact of slightly increased radiation levels in the plant against the equipment qualification criteria for systems and components located in the affected areas. Any changes in qualified life or service will be accounted for in the design/installation of the HWC system and reflected in [on] the plant prior to HWC system operation.

With regard to the presence of hydrogen and oxygen in the yard, the two mediums meet the requirements of NPFA 50 and 50A for separation from the facility as discussed in Item III.3 above [of the licensee's submittal]. UFSAR Section 9.5.1.1.11 has analyzed the presence and storage of combustible materials in the yard and the HWC hydrogen and oxygen storage facilities do not affect the conclusions reached (other than the incorporation of the storage information in Table 9.5-3). Finally, in following the EPRI Guidelines and addressing the NRC staff requirements in Item III above, PSE&G concludes that the probability for an explosion, flammable vapor cloud or fire is minimized. Even if such an accident were to occur, there would be no impact of the station due to the separation distance to the storage vessels from safety related structures. Thus the information contained in UFSAR Section 2.2.3.1 is not affected due to the presence of a HWC system.

Finally, extensive safety features for the HWC system have been established which provide assurance that the operation of the system at HCGS will not create an unacceptable situation nor adversely impact the operation of any other system. Therefore, since the changes to the Technical Specifications themselves do not affect existing system function nor create a situation which has not been previously analyzed and appropriately designed for, the changes do not create any new or different kinds of accidents than previously evaluated.

3. Do not involve a significant reduction in a margin of safety.

The proposed changes to the Technical Specifications contain specific requirements regarding their applicability:

- Operation of the HWC system is only permitted above 20 percent of Rated Thermal Power.
- When the HWC system is in operation the MSL radiation monitor setpoints can be adjusted upward, to levels previously determined during the hydrogen injection test, to account for the increase in the background MSL radiation levels.
- Prior to decreasing reactor power to below 20% of Rated Thermal Power, the setpoints must be readjusted to their pre-HWC system operation levels.
- If the power level falls below 20% without the setpoint change, control rod motion is suspended (except for scrams or other

emergency situations) until the setpoint adjustment is made.

These requirements will assure that the HWC system is operated safely and with sufficient margin such that spurious MSL isolations are precluded while still assuring that any gross failures in the fuel cladding remain detectable.

As discussed in Item IV.1 above [of the licensee's submittal], the CRDA is the only accident which takes credit for the MSL isolation trip function; however, above 20 percent power, the consequences of the CRDA are so minimal that they may be considered negligible (reference the above cited NEDO report.) Therefore, the change in the Technical Specification setpoint has no significant effect on the margins of safety for this accident scenerio and the restriction regarding suspending control rod motion further assures that during setpoint adjustments, a CRDA is minimized.

Finally as discussed in Item III.6 above [of the licensee's submittal], the increase in background radiation levels has been analyzed and PSE&G has concluded that neither plant personnel nor the health and safety of the public are at risk when operating with the HWC system. Therefore, it can be concluded that the proposed changes do not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Walter R. Butler

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

Date of amendment request: September 28, 1988

Description of amendment request: This amendment would revise Technical Specification Table 3.3.7.5-1 to permit actions consistent with Technical Specification 3.6.3 regarding allowable out-of-service times for inoperable primary containment isolation valves and their associated position indication instrumentation. The change would avert the currently required plant shutdown in the event that position indication instrumentation for a primary containment isolation valve in an otherwise isolated penetration is declared inoperable.

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

The proposed change to the HCGS Technical Specifications:

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

As required by Action (a) of Technical Specification 3.6.3, should a primary containment isolation valve be declared inoperable the affected penetration must be isolated. This isolation can be accomplished by either deactivating at least one automatic valve, closing at least one manual valve or installing a blind flange in the affected penetration. Furthermore, the system for which the inoperable valve provides containment isolation must also be declared inoperable and the appropriate Action statements for that system performed. Assuming that the plant can continue to operate under these conditions, the concern which must be addressed as a result of this proposed change is whether or not the probability or consequences of an accident previously evaluated are significantly increased when the position indication instrumentation for an otherwise inoperable containment isolation valve is permitted to remain inoperable longer than the currently imposed 30 or 7 days, per Action 82(a) and 82(b) of Technical Specification Table 3.3.7.5-1, respectively.

The requirement to isolate the affected penetration due to an inoperable valve establishes containment isolation for that penetration. This action establishes a safe configuration for continued operation assuming of course that the affected system is not required to remain Operable. For those systems which can be isolated without jeopardizing continued safe operation, the need for monitoring containment isolation is no longer necessary as isolation has already been achieved.

Therefore, it can be concluded that if the provisions of Technical Specification 3.6.3, Action a.2 or a.3 are in effect:

(i) the penetration is in a safe configuration with regards to the provisions for containment isolation - closed,

(ii) spurious movement of the valve is precluded by either the lack of power, the need for local manual operation, or the presence of an installed blind flange, and

(iii) administrative controls and surveillance requirements exist to assure continued containment isolation.

The current requirements of ACTION 82 of Technical Specification Table 3.3.7.5-1 regarding AOT for primary containment isolation valve position indication instrumentation serve no purpose with regard to assurance of containment integrity if and only if the associated penetration is isolated pursuant to Technical Specification 3.6.3, Action a.2 or a.3. This function is adequately controlled under the auspices of Technical Specifications 4.6.1.1 and the administrative controls already in place. Consequently, extending the AOT for inoperable position indication in penetrations isolated as described above does not represent an increase in the probability or consequences of a previously evaluated accident since containment isolation (the accident function of concern) is already achieved and assured.

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

The proposed change does not negate the requirement for containment integrity or isolation but simply makes use of the existing Technical Specifications which require such. By (sic) extending the provisions for isolated penetrations to the required actions for associated inoperable position indication simply takes advantage of the physical constraints and administrative controls already imposed.

Furthermore, the proposed change does not require any plant modification nor (sic) design change but merely permits a specific case of inoperability to exist while plant operation continues. This condition is well bounded in terms of the extent to which inoperability is permitted. Additionally, the flexibility provided by this proposed change will not result in a change to the operational characteristic of any system or process. The inoperability of primary containment isolation valve position indication instrumentation is already permitted for the currently identified AOT. This change simply extends the AOT as long as other compensating measures are in effect.

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

(3) Does not involve a significant reduction in a margin of safety.

The increase in AOT for an inoperable primary containment isolation valve position indication instrument from either 7 or 30 days to an unlimited time does not decrease the margin of safety since a more restrictive compensating measure is in effect, namely the subject penetration is in the safe configuration with regard to containment isolation provisions - closed. Therefore, the margin of safety remains the same as that permitted by Technical Specification 3.6.3, Action a.2 or a.3. With the penetration in an isolated position and the assurances available that such a position will be maintained, the maximum margin of safety

has been achieved, i.e. the penetration has been placed in the post accident configuration.

The length of time that either or both position indication instrumentation channels for either or both containment isolation valves remain inoperable has no bearing on the position of the valves in the subject penetration. Realizing that the information provided by the position indication instrumentation is simply indication only, i.e. no automatic isolation or actuation function results from loss of or change in position indication, further substantiates the proposed change. Therefore, it can be concluded that the proposed change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Walter R. Butler

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

Date of amendment request: September 19, 1988 as supplemented November 4, 1988.

Description of amendment request: The proposed changes would delete the requirement to perform local leak rate tests (LLRT) on containment penetration pipes associated with two systems, high pressure injection and decay heat removal systems. The licensee contends that the penetrations being removed from the LLRT requirements would be filled with water and pressurized to a pressure greater than the maximum containment pressure associated with accident conditions. As a result, these penetrations would not provide a pathway for radioactive contaminants in the containment atmosphere to escape to the environment.

The proposed change would also increase the interval between LLRT's from 18 months to a maximum of 24 months.

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 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: (1) involve a significant increase in the probability or consequences of an accident previously evaluated because based on the licensee's evaluation, the containment penetrations being deleted from the LLRT program are not potential release pathways for airborne activity during accident conditions and therefore will not change accident consequences; increasing the LLRT interval from 18 months to 24 months is in accordance with regulatory guidance and is not a significant change in terms of accident consequences; (2) create the possibility of a new or different kind of accident from any accident previously evaluated because containment penetrations and LLRT's are an integral part of accident evaluations, and the proposed changes do not create new or different accident concerns; (3) involve a significant reduction in a margin of safety. The proposed changes are relatively minor changes to the LLRT program. The penetrations being deleted from the LLRT requirements are a small fraction of all containment penetrations and even under worst conditions, radioactive releases through these penetrations would constitute a small fraction of releases from all pathways.

Based on the above discussion, the staff proposes to determine that the proposed amendment does 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, P. O. Box 15830, Sacramento, California 95813.

NRC Project Director: George W. Knighton

Virginia Electric and Power Company, Docket No. 50-338, North Anna Power Station, Unit No. 1, Louisa County, Virginia

Date of amendment request: October 19, 1988

Description of amendment request: The proposed change involves an amendment, in the form of a license condition, to Operating License No. NPF-4 for NA-1. Specifically, the proposed license condition allows a one-time extension of the surveillance test intervals for certain surveillance tests as specified in the NA-1 Technical Specifications (TS) for the seventh cycle of operation. NA-1 completed applicable Mode 4, 5 and 6 surveillance tests during the sixth refueling outage which ended on June 29, 1987. It was not considered reasonable to repeat these surveillance tests during the time frame that Unit 1 was shutdown for steam generator repairs which occurred from July 15, 1987 to October 13, 1987. However, this unplanned outage did serve to impact the surveillance test intervals between the sixth and the forthcoming seventh refueling outages. This delay, together with additional time allowed for an optimum fuel burn-up before the next refueling, has resulted in a deferral of the next refueling outage for NA-1 until April 1989.

Currently, NA-1 TS require the performance of certain surveillance tests at 18, 36, and 60 month intervals to coincide with normal 18-month refueling cycles.

The proposed change would extend these surveillance test intervals for the NA-1 seventh cycle by 6 months to compensate for several unanticipated outages including the steam generator tube rupture event and to permit optimum fuel burn-up prior to refueling. Use of the allowable extension of the surveillance intervals in accordance with Specification 4.0.2 of the TS would require an extension corresponding to the 3-month unplanned outage. Rather than use the extension allowed by Specification 4.0.2 and request an additional extension, a 6-month extension for the affected surveillance test intervals is requested for the seventh cycle only to preserve the extension allowed by Specification 4.0.2 for future refueling cycles.

One-time changes to the surveillance test intervals associated with a plant shutdown or refueling outage as specified in the TS for License Number NPF-4 are requested as follows:

(1) The 18-month surveillance test cycle requirement as specified in the following TS sections would be changed to 24 months for the seventh cycle of unit operation only:

4.1.2.2.c	4.3.2.1.2	4.3.3.9.c	4.4.3.2.1.b
4.9.3.1.b	4.4.10.1.1	4.5.1.d	4.5.2.e
4.6.1.3.c	4.6.2.1.c	4.6.2.2.1.c	4.6.2.3.c
4.6.3.1.2	4.7.1.2.b	4.7.4.1.c	4.7.7.1.d
4.7.8.1.d	4.7.9.1.b	4.7.10.c	4.7.10.f
4.8.1.1.1.b	4.8.1.1.2.d	4.8.1.1.3.e	4.8.2.3.2.d
4.8.2.3.2.f			

(2) The 18/36-month surveillance test cycle requirement as specified in the following TS sections would be changed to 24/42 months for the seventh cycle of unit operation only:

4.3.1.1.3 4.3.2.1.3

(3) The 60-month surveillance test interval requirement as specified in the following TS section would be changed to 66 months for the seventh cycle of unit operation only:

4.8.1.1.3.d

(4) Table 1.2 of Section 1.0, Definitions, which defines "R" as "At least once per 18 months" as it applies to the following TS sections and related tables, and the 18-month requirement in the note in the tables indicated below, would be changed to 24 months for the seventh cycle of unit operation only:

Section	Table	Note
4.3.1.1.1	4.3-1	(4)
4.3.2.1.1	4.3-2	(1)
4.3.3.1	4.3-3	—
4.3.3.3.1	4.3-4	—
4.3.3.5	4.3-6	—
4.3.3.6	4.3-7	—

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

The licensee has evaluated the change request against the standards provided above and has determined that this change will not:

Involve a significant increase in the probability or consequences of an accident previously evaluated. Current monitoring instrumentation and ongoing [TS] surveillance tests ensure the equipment and systems involved in the extended surveillance interval will remain in an operable condition until their inspection at the next refueling outage.

Create the possibility of a new or different kind of accident from any accident previously evaluated. Extending the interval for the performance of specific surveillance tests does not create the possibility of a new or different kind of accident. Periodic surveillance tests have been performed since the sixth outage to monitor system and component performance and to detect degradation. Surveillance tests will continue to be performed during the extension interval.

Involve a significant reduction in the margin of safety. Extending the interval for these specific surveillance tests for the [seventh] cycle of [NA-1] does not significantly degrade the margin of safety. Surveillance tests will continue to be performed during the extension interval. Current monitoring instrumentation and ongoing [TS] surveillance tests ensure the affected equipment and systems remain in an operable condition.

The NRC staff has made a preliminary review of the licensee's analyses of the proposed change and agrees with the licensee's conclusion that the three standards in 10 CFR 50.92(c) are met. Therefore, the staff proposes to determine that the proposed amendment does not involve a significant hazards consideration.

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

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, P.O. Box 1535, Richmond, Virginia 23212.

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: July 20, 1988

Description of amendment request: The proposed change would clarify the current NA-1&2 Technical Specifications (TS) regarding reactor coolant system leakage detection systems. Specifically, the change would clarify the NA-1&2 TS 3.4.6.1 regarding reactor coolant system leakage detection systems and bring the present TS into closer agreement with Regulatory Guide 1.45 and Revision 4 of the Westinghouse Standard TS which are appropriate to NA-1&2.

The current TS Limiting Condition of Operation (LCO) is difficult to understand and can be interpreted to require two leakage detection systems to be operable, whereas the associated action statement can be interpreted to require three separate and independent methods to be operable. Regulatory Guide 1.45 requires three separate detection methods of which two of the methods should be: (1) the containment particulate radioactivity monitoring system and (2) the containment sump level and discharge flow measurement system. Regulatory Guide 1.45 also requires a third method which is satisfied by the containment gaseous radioactivity monitoring system.

The proposed change would clarify the TS such that the containment particulate and gaseous monitoring system are considered as two separate detection methods but are not considered as two independent systems. Specifically, the monitors share a common piping system, power supply and piping arrangement that do not make them truly independent. Therefore, the action statement would be modified to achieve consistency with the LCO. Specifically, if either of the two required leakage monitoring systems are inoperable, a compensatory leakage measurement using the RCS water inventory balance method would be specified instead of obtaining grab samples. The current TS does not require a compensatory leakage measurement if the containment sump discharge measurement system is inoperable whereas the revised TS does. This compensatory leakage measurement along with a fully operable leakage detection system is the basis for extending the action statement from 6 hours to 30 days when one leakage detection system is inoperable. The surveillance requirements have also been rewritten to require a periodic calibration of the containment sump level monitor.

The proposed TS changes are consistent with the regulatory position of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems" and NUREG-0452, "Westinghouse Standard Technical Specifications." Specifically, three separate detection methods are

provided but they are grouped as two separate and redundant detection systems. The loss of a single system would not result in the loss of detection capability. Therefore, regulatory position 9 of Regulatory Guide 1.45 is fully met.

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

The licensee has evaluated the change request against the standards provided above and has determined that the proposed changes would not:

(a) result in a significant increase in the probability or consequence of an accident previously evaluated, i.e., a loss of coolant accident, because the specification continues to require two redundant and diverse means of continuously monitoring for reactor coolant system leakage. In addition, an operability requirement for the containment sump level monitor has been added to the LCO, and a requirement to implement a compensatory leakage measurement (i.e., inventory mass balance) if either or both of the sump leakage monitors are inoperable has been added to the Action Statement. Finally, the requirement to obtain and analyze appropriate containment grab samples if one of the radioactivity monitors is inoperable has been replaced with a requirement to perform a compensatory leakage measurement using the mass balance method. This method is considered equivalent to the grab sample method in terms of leakage detection sensitivity and therefore will provide the same level of protection as previously provided.

(b) create the possibility of a new or different kind of accident. The additional required reactor coolant system leakage monitor (i.e., sump level) is already required by the related [TS] 3/4.4.6.2 regarding reactor coolant system leakage limits (see [TS] 4.4.6.2.1b) and therefore does not introduce any new or unique accident precursors. Similarly, the additional required compensatory leakage measurement (i.e., inventory mass balance) is a test that is routinely performed in accordance with [TS] 4.4.6.2.1d and existing station period test procedures, and therefore does not create any new or unique accident precursors.

(c) result in a significant reduction in the margins of safety as defined in the bases for any [TS] because the proposed [TS] continue to require two redundant and diverse means of leakage monitoring as well as a compensatory leakage measurement every 24 hours if either of the two required leakage monitoring systems is inoperable, and therefore the [TS] Bases and the recommendations of Regulatory Guide 1.45 (regulatory position 9 regarding [TS]) continue to be satisfied.

The NRC staff has made a preliminary review of the licensee's analyses of the proposed change and agrees with the licensee's conclusion that the three standards in 10 CFR 50.92(c) are met. Therefore, the staff proposes to determine that the proposed amendments do not involve a significant hazards consideration.

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

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, P.O. Box 1535, Richmond, Virginia 23212.

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: September 30, 1988

Description of amendment request: The proposed change would support full power operation for NA-1&2 at steam generator (SG) tube plugging levels of up to 18%. The results of the analysis supporting the increase in SG tube plugging limits also support a new maximum core peaking factor (FQ) of 2.19. To support the proposed change a reanalysis of the Emergency Core Cooling System (ECCS) performance for the postulated large-break loss-of-coolant accident (LOCA) has been performed in compliance with Appendix K to 10 CFR Part 50. The results of the reanalysis are presented in compliance with 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors." The analysis was performed with the NRC-approved 1981 model with BART version of the Westinghouse LOCA-ECCS evaluation model. The analysis includes the evaluation model revisions described in WCAP-9561-P, Addendum 3, Revision 1, "Addendum to BART-A1: A Computer Code for the Best Estimate Analysis of Reload Transients," dated July 1986 and approved by NRC letter dated August 25, 1988. The analytical techniques used are in full compliance with 10 CFR Part 50 Appendix K.

As required by Appendix K of 10 CFR Part 50, certain conservative assumptions were made for the LOCA-ECCS analysis. The assumptions pertain to the conditions of the NA-1&2 reactors and associated safety system equipment at the time that the LOCA is assumed to occur. These assumptions include such items as the core peaking factors, the containment pressure, and the performance of the ECCS. All assumptions and initial operating conditions used in this reanalysis were the same as those used in previous LOCA-ECCS analyses, with two exceptions. The steam generator plugging level was increased to 16% (from 7% and 15%) and the maximum core peaking factor, FQ, was increased from 2.15 to 2.19. With these changes incorporated into the analysis, it was found that the LOCA analysis results continue to meet the 10 CFR 50.46 acceptance criteria.

The large-break LOCA transient is divided, for analytical purposes, into three phases: blowdown, refill, and reflood. There are three distinct transients analyzed in each phase, including the thermal-hydraulic transient in the reactor coolant system, the pressure and temperature transient within the containment and the fuel clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of interrelated computer codes were used for the analysis.

The description of the various aspects of the LOCA analysis methodology is provided in WCAP-8339, "Westinghouse ECCS Evaluation Model-Summary," dated July 1974. This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of codes that ensure compliance with 10 CFR Part 50, Appendix K. The SATAN-VI, COCO, WREFLOOD, BART, and LOCTA-IV codes, which are used in the LOCA analysis, are described in detail in WCAP-8306, WCAP-8326, WCAP-8171, WCAP-9695, WCAP-10062 and WCAP-8305, respectively. These codes assess whether sufficient heat transfer geometry and core amenability to cooling are preserved during the time spans applicable to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient in the reactor coolant system (RCS) during blowdown, and the COCO computer code calculates the containment pressure transient during all three phases of the LOCA analysis. The thermal-hydraulic response of the RCS during refill and reflood is calculated by the WREFLOOD computer code. A mechanistic estimate of the heat transfer coefficient in the core during reflood is provided by the BART computer code. For the three phases of the LOCA, the

LOCTA-IV computer code is used to compute the thermal transient of the hottest fuel rod.

SATAN-VI is used to determine the RCS pressure, enthalpy, and density, as well as the mass and energy flow rates in the RCS and steam generator secondary side, as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator mass and pressure and the pipe break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown, the mass and energy release rates during blowdown are transferred to the COCO code for use in the determination of the containment pressure response during the first phase of the LOCA. Additional SATAN-VI output data from the end of the blowdown, including the core inlet flowrate and enthalpy, the core pressure, and the core power decay transient, are input to the LOCTA-IV code.

With input from the SATAN-VI code, WREFLOOD uses a system thermal-hydraulic model to determine the core flooding rate (i.e., the rate at which coolant enters the bottom of the core), the coolant pressure and temperature, and the quench front height during the refill and reflood phases of the LOCA. WREFLOOD also calculates the mass and energy flow rates that are assumed to be vented to the containment. Since the mass flowrate to the containment depends upon the core pressure, which is a function of the containment backpressure, the WREFLOOD and COCO codes are interactively linked. With the input and boundary conditions from WREFLOOD, the mechanistic core heat transfer model in BART calculates the fluid and heat transfer conditions in the core during reflood.

LOCTA-IV is used throughout the analysis of the LOCA transient to calculate the fuel and clad temperatures of the hottest rod in the core. The input to LOCTA-IV consists of appropriate thermal-hydraulic outputs from SATAN-VI, WREFLOOD and BART, and conservatively selected initial RCS operating conditions.

The COCO code, which is also used throughout the LOCA analysis, calculates the containment pressure. Input to COCO is obtained from the mass and energy flowrates assumed to be vented to the containment, as calculated by the SATAN-VI and WREFLOOD codes. In addition, conservatively chosen initial containment conditions and an assumed mode of operation for the containment cooling system are input to COCO.

The NA-1&2 LOCA-ECCS reanalysis has evaluated plant operation at SG tube plugging levels of up to 18% based on the acceptance criteria delineated in 10 CFR 50.46. The evaluation concluded that reanalysis of non-LOCA accidents is not required to support this increased tube plugging level provided the measured RCS flow rate remains above the thermal design flow rate assumed for the safety analyses. SG tube plugging in sufficient quantity can potentially affect non-LOCA safety analysis due to reduced primary system flow, more severe pump coastdown characteristics, and the reduction of the reactor primary coolant system volume. Primary flowrate becomes a key parameter in the Departure from Nucleate Boiling Ratio (DNBR) limited events (e.g., uncontrolled RCCA bank withdrawal at power) when it falls below the thermal design flowrate. Pump coastdown characteristics impact analysis results when they become more severe than the conservative values used in the loss-of-flow related analyses. The reduced primary coolant system volume affects dilution times in uncontrolled boron dilution events.

A conservative estimate of the NA-1&2 RCS flow versus tube plugging is based on past flow measurements taken at NA-1&2 for several levels of steam generator tube plugging. More recent NA-1 measurements at greater tube plugging levels validate the conservatism of RCS flow versus tube plugging curve. A re-evaluation of the projection indicates that the conservatively estimated flow rate at the proposed 18% plugging level is approximately equal to the North Anna thermal design flow. Therefore, while measured flow exceeds the thermal design flow, the current docketed licensing analyses remain valid for those events in which flow rate is an important concern.

The impact of 18% tube plugging on dilution times in the uncontrolled boron dilution events was also evaluated. Relative to the boron dilution events, the evaluation indicated: (1) for uncontrolled dilution during startup, time to criticality is 37 minutes. This is more than adequate time for the operator to recognize the high count rate signal and terminate the dilution flow, and (2) for uncontrolled dilution at power, the operator has ample time (greater than 15 minutes) after the over-temperature delta T alarm or trip to determine the cause of dilution, isolate the water source, and initiate reboration before total shutdown margin is lost due to dilution.

Tube plugging levels exhibit no influence on dilution times for the refueling mode of operation, since the SG volumes are not a part of the active system. The evaluation shows that for SG tube plugging levels of up to 18 percent, no reanalysis of the DNBR related non-LOCA safety events is necessary and that the currently licensed analyses remain valid. In the case of the uncontrolled boron dilution events, the available operator response times for the startup and at power evaluations are reduced but remain well above the minimum acceptance values.

Based on the large break LOCA analysis, a double-ended cold-leg guillotine break with a discharge coefficient (C_D) of 0.4 was found to be the limiting break size and location. The analysis resulted in a limiting peak clad temperature of 2165.2° F for the $C_D = 0.4$ case, a maximum local cladding oxidation level of 5.77%, and a total core metal-water reaction of less than 0.3%.

For breaks up to and including the double-ended rupture of a reactor coolant pipe, the ECCS will meet the acceptance criteria as presented in 10 CFR 50.46, as follows: (1) the calculated peak fuel rod clad temperature is below the requirement of 2200° F, (2) the amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of Zircaloy in the reactor, (3) the clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limits of 17% are not exceeded during or after quenching, (4) the core remains amenable to cooling during and after the break, and (5) the core temperature is reduced and the long-term decay heat is removed for an extended period of time.

The effects of increasing the allowable steam generator tube plugging to 18% has been assessed for existing non-LOCA event analyses. This evaluation has concluded: (1) current analyses for which RCS flow is an important concern remain valid as long as measured flow is greater than the thermal design flow assumed in safety analyses, (2) the existing loss-of-flow related analyses assume a conservative reactor coolant pump flow coastdown characteristic which accommodates the effect of increased tube plugging on loop flow resistance, and (3) boron dilution analyses assuming the reduced RCS volume associated with tube plugging result in dilution times which remain adequate for the required operator actions to be performed.

Basis for proposed no significant hazards consideration determination:

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

The licensee has evaluated the change request against the standards provided above and has determined that:

1. Since the proposed changes involve parameters which are not accident initiators, they will not increase the probability of occurrence of any malfunction or accident previously addressed. The reanalyzed large break LOCA analysis verifies that operation under the revised specifications would also not result in any increase in accident consequences over those in previously accepted analyses.

2. No new accident types or equipment malfunction scenarios will be introduced as a result of operating in accordance with the revised specifications. The change which potentially affects physical components in the plant systems (steam generator tube plugging) was explicitly included in the analysis and shown not to produce any new or unique accident precursors.

3. The margin of safety, as defined in the basis for the plant Technical Specifications, is not reduced. The revised ECCS analysis meets the acceptance criteria of 10 CFR 50.46. Additionally, since evaluation of non-LOCA accidents concluded that acceptance criteria are met when considering the proposed changes, the current margin of safety is maintained for LOCA and non-LOCA accidents.

Based on the above evaluation, the licensee has determined that the proposed change involves no significant hazards considerations.

The NRC staff has made a preliminary review of the licensee's analyses of the proposed change and agrees with the licensee's conclusion that the three standards in 10 CFR 50.92(c) are met. Therefore, the staff proposes to determine that the proposed amendments do not involve a significant hazards consideration.

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

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, P.O. Box 1535, Richmond, Virginia 23212.

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: September 30, 1988

Description of amendment request: The proposed change would increase the allowable enrichment of fuel assemblies irradiated at NA-1&2 to 4.3 weight percent (w/o) U-235. An increase in the current NA-1&2 Technical Specifications (TS) limit of 4.1 w/o U-235 to 4.3 w/o U-235 would allow an increase in batch average discharge burnup to levels approaching the currently licensed limit of 45,000 Mega-Watt Days per Metric Ton Uranium (MWD/MTU). The enrichments currently used limit the batch average burnup to a value from 38,000 MWD/MTU to 42,000 MWD/MTU depending on the number of fuel assemblies loaded each cycle. An increase in the enrichment limit would result in significant fuel cycle cost savings and enhance fuel management plans to increase batch average discharge burnups.

The safety impact for operation of NA-1&2 with high burnup fuel was previously addressed by the licensee in letters to the NRC dated December 4, 1980, March 6 and 26, 1981 and July 24, 1981. By letter dated April 9, 1984, the NRC approved operation of NA-1&2 to a batch of discharge of 45,000 MWD/MTU. A generic impact of extended burnup on the design and operation of Westinghouse fuel was addressed in WCAP-10125-P-A, "Extended Burnup Evaluation of Westinghouse Fuel," dated December 1985. In addition, the NRC made an independent assessment of the environmental and economic impacts of the use of extended burnup fuel in light water power reactors. This assessment was dated February 1988 and entitled "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," Pacific Northwest Laboratory, NUREG/CR-3009. The overall findings of NUREG/CR-3009 were that no significant adverse effects would be generated by increasing the present batch-average burnup level to values of 50,000 MWD/MTU or above, as long as the maximum rod average burnup of any rod is no greater than 60,000 MWD/MTU. Since the findings of these evaluations provided in NUREG/CR-3009 concerning the impact of extended burnup fuel are valid for an enrichment of 4.3 w/o U-235, and since the NA-1&2 spent fuel storage facility is currently licensed to 4.3 w/o U-235, the license's submittal addresses only the impact of increased enrichment on the requirements for the currently approved new fuel storage racks at NA-1&2.

The specific 10 CFR Part 50 Appendix A General Design Criteria for new fuel storage facilities are listed in Section 9.1.1 of the Standard Review Plan (NUREG-0800). Since no physical modifications are being made to the current NA-1&2 new fuel racks, the licensee's analysis only addresses the impact of the increased enrichment on the requirement of subcriticality under normal and postulated abnormal rack conditions (General Design Criterion 62). The highest K-effective allowable by Section 9.1.1 of NUREG-0800 for all conditions is 0.98.

The computer modeling of the storage racks was performed in three-dimensions (3-D) to minimize unnecessary conservatism and uncertainty. All K-effective calculations were performed with the Monte-Carlo program KENO V.a and contained within the modular code system SCALE. KENO V.a is addressed in ORNL-NUREG-CSD-2-VI-R2 entitled "KENO V.a, An Improved Monte Carlo Criticality Program with Supergrouping," dated December 1984. SCALE is addressed in ORNL-NUREG-CSD-2-VI-R3, "SCALE: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation," dated December 1984. The SCALE package automatically processes cross sections to create a set of resonance self shielded cross sections for use by KENO. Because all calculations for this analysis were made using a discrete pin representation, no spatial self shielding was performed prior to the KENO execution. The cross section set chosen was the 27 group ENDF/B-IV data contained in the SCALE package. Sufficient neutron histories were run for each case to limit the statistical uncertainties in the K-effective to less than 0.4% delta K/K.

The results of the licensee's analysis indicate that for a fuel enrichment of 4.3 w/o U-235, the NA-1&2 fuel storage area meets the criticality limit of K-effective less than 0.98 and is safe under the criticality specifications set forth in the NRC Standard Review Plan (NUREG-0800).

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 change does not involve a significant hazards consideration because operation of NA-1&2 in accordance with the proposed change would not:

1. Involve a significant increase in the probability or consequences of accidents previously evaluated. The only accident scenarios for which the probability of occurrence are potentially affected by fuel enrichments involve criticality events during fuel handling and storage. The criticality safety analyses demonstrates that K-effective during fuel handling and storage of new fuel is low enough to ensure subcriticality during postulated accident conditions. The probability of occurrence of criticality during fuel handling or storage is therefore not increased. Since subcriticality is maintained, no releases would result from the fuel handling and storage accident scenarios. In addition, since the burnup limit will not be increased beyond that already approved in NRC letter dated April 9, 1984, the radiological consequences of the accidents discussed in WCAP-10125-P-A and NUREG/CR-3009 will not be increased.

2. The proposed amendments do not create the possibility of a new or different kind of accident from any previously evaluated. The only potential impact of increased enrichment upon fuel storage and handling involves the potential for criticality and the licensee's analyses that has determined that subcriticality will be maintained.

3. The proposed amendments do not involve a significant reduction in the margin of safety. The criticality analysis demonstrates that there is adequate margin to ensure subcriticality of the fuel during storage and handling of new fuel. The NRC safety analysis provided in a letter dated December 21, 1984, provides the same assurance for spent fuel.

Therefore, pursuant to 10 CFR 50.92, based on the above considerations, it has been determined that these changes do not constitute a significant safety hazards consideration.

The NRC staff has made a preliminary review of the proposed change and concludes that the three standards in 10 CFR 50.92(c) are met. Therefore, the staff proposes to determine that the proposed amendments do not involve a significant hazards consideration.

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

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, P.O. Box 1535, Richmond, Virginia 23212.

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: September 30, 1988

Description of amendment request: The proposed change would allow the direct reactor trip on turbine trip to be blocked below 30% of the rated thermal power. Currently, Permissive Setpoint P-7 is used to block the reactor trip on turbine trip below 10% of rated thermal power. The proposed modification would rewire the Solid State Protection System so that Permissive Setpoint P-8 is used to block the reactor trip on turbine trip below 30% power. A licensee review of historical trip data shows that the most commonly occurring reactor trip on turbine trip events are well below 30% of rated thermal power. Thus, it was concluded that the use of the existing P-8 bistable to block the direct reactor trip on turbine trip would be an effective means of reducing unneeded trips at low power. Direct reactor trip on turbine trip would be available above 30% power. The plant's designed load rejection capability is 50% of full load.

At present, for all power levels above 10% (the P-7 permissive setpoint) of Rated Thermal Power (RTP), the NA-1&2 nuclear reactors are tripped directly on turbine trip from a signal derived from the turbine autostop oil pressure or turbine stop valve position. A direct reactor/turbine trip at low power is unnecessary and unduly stresses plant systems. Thus, the licensee is proposing a change which would allow for a block of the direct reactor trip on turbine trip below 30% of rated thermal power.

The proposed modification would rewire the Solid State Protection System so that Permissive P-8 is also used to block the reactor trip on turbine trip instead of Permissive P-7. It was concluded that the use of the existing P-8 bistable to block the direct reactor trip on turbine trip would be an effective means of eliminating unneeded low power transient reactor trips. Direct reactor trip on turbine trip would still be available above 30% power.

Three items were considered in the licensee's safety analysis and were addressed in the licensee's submittal.

(1) The results of the worst-case analyses show that a total loss of external electrical load without a direct or immediate reactor trip below 30% of RTP presents no hazards to the integrity of the reactor coolant system or the main steam system. Pressure-relieving devices incorporated in the two systems are adequate to keep the maximum pressure within the design limits. The licensee concluded that the results of this analysis demonstrates that plant parameters are maintained within the design limits previously analyzed for the loss-of-load accident from full power described in Section 15.2.7 of the NA-1&2 Updated Final Safety Analysis Report (UFSAR).

(2) An analysis was also performed for a complete loss of forced reactor coolant flow initiated from the most adverse preconditions of a turbine trip, and demonstrated that the integrity of the core is maintained by operation of the reactor protection system, i.e., the DNBR will be maintained above the design limit value. Thus, there will be no cladding damage and no release of fission products to the reactor coolant system. The licensee concluded that plant parameters are maintained within design limits and that this analysis is bounded by the results of a complete loss of flow from full power as described in Section 15.3.4 of the UFSAR.

(3) Finally, an analysis was conducted to verify that the applicable NUREG-0737 requirements were met. NUREG-0737 required that the frequency of a small break loss-of-coolant-accident (LOCA) caused by a stuck-open pressurizer power operated relief valve (PORV) be reduced and that it be demonstrated not to be a significant contributor to the probability of a small break LOCA. Both the loss-of-load and the loss-of-flow accidents have the potential of causing the PORV to open. The licensee has conducted an analysis which demonstrates that the PORVs are not normally challenged during this event and thus the NUREG-0737 requirements are met.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from

any accident previously evaluated: or (3) involve a significant reduction in a margin of safety.

The licensee has evaluated the change request against the standards provided above and has determined that:

1. No significant increase in the probability of occurrence or consequences of an accident analyzed in the UFSAR will result from elimination of reactor trip on turbine trip below 30% of Rated Thermal Power (RTP). The analyses results show that the DNBR does not decrease below the design limit at any time. The analysis also shows that, except under [most] conservative assumptions, the pressurizer PORVs are not challenged during the transient. Pressure-relieving devices incorporated in the primary and the secondary systems are adequate to keep the maximum pressure within the design limit. Since the predicted results are within the range of existing safety analysis values, it is concluded that operation with the proposed Technical Specification changes will neither significantly increase the probability of occurrence nor the consequences of initiating events for any known accident.

2. No new or different accident type not previously considered in the UFSAR is created by this proposed change. The complete loss of unit load without a direct reactor trip on turbine trip is a design event and is addressed in Section 15.2.7 of the UFSAR. The results for a loss of flow due to fast bus transfer failure after a turbine trip are bounded by the results for a complete loss of flow from full power, which is discussed in Section 15.3.4 of the UFSAR. Thus, the results of all the relevant accident analyses show that operation with this modification does not create a new or different accident type than any evaluated previously in the UFSAR.

3. The margin of safety is not reduced. The proposed Technical Specification changes have been incorporated in the safety analyses. These analyses have demonstrated that calculated results meet all design acceptance criteria as stated in the UFSAR.

Based on the above evaluation, the licensee has determined that the proposed change involves no significant hazards considerations.

The NRC staff has made a preliminary review of the licensee's analyses of the proposed change and agrees with the licensee's conclusion that the three standards in 10 CFR 50.92(c) are met. Therefore, the staff proposes to determine that the proposed amendments do not involve a significant hazards consideration.

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

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, P.O. Box 1535, Richmond, Virginia 23212.

NRC Project Director: Herbert N. Berkow

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

Date of amendments request: January 6, 1987 as supplemented April 14 and May 15, 1987.

Description of amendments request: The licensee proposes to change Technical Specification Table 15.4.1-1, "Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels," to increase the period of the logic channel test of the reactor trip on low reactor coolant flow in both loops from monthly to each (annual) refueling outage.

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.

In 51 FR 7751, the Commission cited examples of amendments that are considered not likely to involve significant hazards considerations. Example (vi) involves a change which may either result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan.

The amendment proposed by the licensee would increase the logic testing period on the two-loop reactor coolant loss of flow reactor trip from monthly to every refueling outage (currently annually). However, the surveillance frequency for the logic channel test of the relays/contacts which initiate the reactor trip on low reactor coolant flow in either loop will remain monthly. Since all bistables and relay coils will still be tested monthly, the net effect of the proposed amendment would be to slightly increase the risk of failing to get a reactor trip upon simultaneous detection of low reactor coolant flow in both loops, due to failure of both of the specific relay contacts which initiate the trip during the increased period between surveillances. Although a numerical quantification of the increase in risk has not been performed, the staff believes that it would be quite small since the event of concern would require two contact failures (one in each loop) between refueling outages and a loss of flow condition occurring while the reactor is between 10% and 50% power.

Section 7.2 of the Standard Review Plan discusses various aspects of the Reactor Trip System. Section 7.2 references Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions." Regulatory Guide 1.22 provides for acceptable methods of testing protection systems during reactor operation. In its May 15, 1987 letter, the licensee states that all actuation devices and all actuated devices which are part of the reactor trip logic for loss of flow in both loops are tested in accordance with the guidance contained in Safety Guide 1.22 (Safety Guide 1.22 and Regulatory Guide 1.22 are identical). Therefore, the proposed change is in conformance with guidance endorsed by the Standard Review Plan.

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

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

NRC Project Director: John N. Hannon.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

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

Notice of Consideration of Issuance of Amendment to Facility Operating License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing in connection with these actions was published in the Federal Register as indicated. No request for a hearing or petition for leave to intervene was filed following this notice.

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

For further details with respect to the action see (1) the applications for amendments, (2) the amendments, and (3) the Commission's related letters, Safety Evaluations and/or Environmental Assessments as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Reactor Projects.

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

Date of application for amendments: March 16, 1988, as supplemented by letter dated July 6, 1988.

Brief description of amendments: The amendments revise Technical Specification Surviellance Requirement 4.5.2.h which specifies flow requirements that the Low Pressure Safety Injection subsystem must meet during flow balance testing.

Date of issuance: October 17, 1988

Effective date: October 17, 1988

Amendment Nos.: 37, 24, and 13

Facility Operating License Nos. NPF-41, NPF-51 and NPF-74: Amendments changed the Technical Specifications.

Date of initial notice in Federal Register: August 10, 1988 (53 FR 30126). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 17, 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 Nos. STN 50-528, STN 50-529 and STN 50-530, Palo Verde Nuclear Generating Station, Units 1, 2 and 3, Maricopa County, Arizona

Date of application for amendments: May 27, 1988

Brief description of amendments: The Amendments revise the technical specification to modify the azimuthal power tilts to require the measured power tilt to be equal to or less than the Core Protection Calculation allowance and the limit in Figure 3.2-1A when the Core Operating Limit Supervisory System is in service. The wording of the surveillance requiment was revised for clarity. In addition, the azimuthal power tilt limit is increased for Unit 2.

Date of issuance: October 17, 1988

Effective date: October 17, 1988

Amendment Nos.: 38, 25, and 14

Facility Operating License Nos. NPF-41, NPF-51 and NPF-74: Amendments changed the Technical Specifications.

Date of initial notice in Federal Register: July 13, 1988 (53 FR 26518). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 17, 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 Nos. STN 50-528, STN 50-529 and STN 50-530, Palo Verde Nuclear Generating Station, Units 1, 2 and 3, Maricopa County, Arizona

Date of application for amendments: September 6, 1988

Brief description of amendments: The amendments revise Technical Specification 6.3.1, "Unit Staff Qualifications," to modify the Senior Reactor Operator license requirements for the Operations Manager.

Date of issuance: October 24, 1988

Effective date: October 24, 1988

Amendment Nos.: 39, 26 and 15

Facility Operating License Nos. NPF-41, NPF-51 and NPF-74: Amendments changed the Technical Specifications.

Date of initial notice in Federal Register: September 21, 1988 (53 FR 36668). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 24, 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 Nos. STN 50-528, STN 50-529 and STN 50-530, Palo Verde Nuclear Generating Station, Units 1, 2 and 3, Maricopa County, Arizona

Date of application for amendments: August 25, 1988, as supplemented by letter dated October 18, 1988.

Brief description of amendments: The amendments delete the organization charts from the technical specifications in accordance with guidance provided by the NRC in Generic Letter 88-06.

Date of issuance: October 25, 1988

Effective date: October 25, 1988

Amendment Nos.: 40, 27 and 16

Facility Operating License Nos. NPF-41, NPF-51 and NPF-74: Amendments changed the Technical Specifications.

Date of initial notice in Federal Register: September 21, 1988 (53 FR 36667). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 25, 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.

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

Date of application for amendment: May 25, 1988

Brief description of amendment: The amendment changes the Technical Specifications to remove the offsite and facility organization charts consistent with the guidance of Generic Letter 88-06, "Removal of Organization Charts from Technical Specifications."

Date of issuance: November 3, 1988

Effective date: November 3, 1988

Amendment No. 120

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

Date of initial notice in Federal Register: June 29, 1988 (53 FR 24506). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 3, 1988.

No significant hazards consideration comments received: No

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

Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units No. 2 and 3, Grundy County, Illinois

Date of application for amendments: August 31, 1988

Brief description of amendments: These amendments modify Section 3.5.F of the Technical Specifications to include more prescriptive requirements for Emergency Core Cooling Systems operability during cold shutdown and refueling operational modes.

Date of issuance: October 26, 1988

Effective date: October 26, 1988 and to be implemented within 60 days.

Amendment Nos.: 101, 97

Provisional Operating License Nos. DPR-19 and DPR-25. These amendments revised the Technical Specifications.

Date of initial notice in Federal Register: September 21, 1988 (53 FR 36669). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated Ocotber 26, 1988.

No significant hazards consideration comments received: No

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

Attorney to licensee: Michael Miller, Esq., Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Daniel R. Muller

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

Date of application for amendments: June 20, 1988

Brief description of amendments: The amendments change Technical Specifications for Dresden Units 2 and 3 to reflect instrumentation enhancements for post-accident monitoring completed per Regulatory Guide 1.97 and NUREG-0737 Supplement 1. In addition several minor corrections and clarifications and have been incorporated.

Date of issuance: November 3, 1988

Effective date: November 3, 1988 and to be implemented within 60 days

Amendment Nos.: 102, 98

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

Date of initial notice in Federal Register: August 10, 1988 (53 FR 30128). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 3, 1988.

No significant hazards consideration comments received: No

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

Connecticut Yankee Atomic Power Company, Docket No. 50-213, Haddam Neck Plant, Middlesex County, Connecticut; Northeast Nuclear Energy Company, et al; Nos. 1, 2, and 3, New London County, Connecticut

Date of application for amendment: April 29, 1988 as supplement by letter dated July 21, 1988.

Brief description of amendment: The changes affect the TSs which specify the qualifications and conduct of the Nuclear Review Board (NRB) for all Units and the Site Nuclear Review Board for Millstone Units 1, 2 and 3.

Date of Issuance: October 26, 1988

Effective date: October 26, 1988

Amendment Nos.: 108, 25, 144, 28

Facility Operating License Nos. DPR-61, DPR-21, DPR-65 and NPF-49. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 24, 1988 (53 FR 32292). The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated October 26, 1988

No significant hazards consideration comments received: No.

Local Public Document Room locations: Russell Library, 123 Broad Street, Middletown, Connecticut 06457 and Waterford Public Library, 49 Rope Ferry Road, Waterford, Connecticut 06385.

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

Date of application for amendment: March 30, 1988 as supplemented April 12, 1988 and September 22, 1988.

Brief description of amendment: The amendment modified Section 3.10 of the Technical Specifications to accommodate the Cycle 12 Core Reload. Specifically, the Minimum Critical Power Ratio (MCPR) and the maximum average planar linear heat generator rated (MAPLHGR) limit was changed. It also permitted the use of GE 8x8EB fuel.

Date of Issuance: October 31, 1988

Effective date: October 31, 1988

Amendment No.: 129

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

Date of initial notice in Federal Register: May 4, 1988 (53 FR 15912). The September 22, 1988 submittal provided additional clarifying information 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 October 31, 1988.

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.

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: June 14, 1988, as supplemented September 27, 1988.

Brief description of amendment: The amendment modified the Technical Specifications to make training requirements be in accordance with 10 CFR 55.

Date of issuance: November 1, 1988

Effective date: November 1, 1988

Amendment No.: 12

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

Date of initial notice in Federal Register: July 13, 1988 (53 FR 26523). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 1, 1988.

No significant hazards consideration comments received: No.

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

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: March 7, 1988, as supplemented April 13, 1988.

Brief description of amendment: This amendment revises Technical Specification 3.1.2 and 4.1.2 for the liquid poison system to comply with the requirements of 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

Date of issuance: October 31, 1988

Effective date: October 31, 1988

Amendment No.: 101

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

Date of initial notice in Federal Register: June 1, 1988 (53 FR 20044). The staff has found one of the requested changes, the revision to Figure 3.1.2b, to be unacceptable and has issued a Notice of Denial. The Commission's related evaluation of the amendment is

contained in a Safety Evaluation dated October 31, 1988.

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.

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

Date of amendment request: July 19, 1988

Brief description of amendment: The amendment modified the Technical Specifications to provide the addition of a Table of Contents for Tables and Figures and to correct an error to a location reference found in Section 2.19.

Date of issuance: November 3, 1988

Effective date: November 3, 1988

Amendment No.: 116

Facility Operating License No. DPR-40. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 24, 1988 (53 FR 32294). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 3, 1988.

No significant hazards consideration comments received: No.

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

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

Date of applications for amendments: November 21, 1986 and November 9, 1987

Brief description of amendments: The amendments modified paragraph 2.E of the licenses to require compliance with the amended Physical Security Plan. This Plan was amended to conform to the requirements of 10 CFR 73.55. Consistent with the provisions of 10 CFR 73.55, search requirements must be implemented within 60 days and miscellaneous amendments within 180 days from the effective date of these amendments.

Date of issuance: October 17, 1988

Effective date: October 17, 1988

Amendment Nos.: 32 and 31

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

Date of initial notice in Federal Register: September 7, 1988 (53 FR 34609). The Commission's related evaluation of the amendments is contained in a Safeguards Evaluation dated October 17, 1988.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: December 18, 1987

Brief description of amendments: Miscellaneous Technical Specification Changes: (a) correct errors; (b) delete redundant information; and (c) change organizational nomenclature.

Date of issuance: October 20, 1988

Effective date: As of the date of issuance.

Amendment Nos.: 83 and 51

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

Date of initial notice in Federal Register: August 10, 1988 (53 FR 30141). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 20, 1988.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: June 3, 1988

Brief description of amendment: The amendment revised the Technical Specifications for material withdrawal schedule and lead factor ratio.

Date of issuance: October 31, 1988

Effective date: October 31, 1988

Amendment No.: 84

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

Date of initial notice in Federal Register: August 10, 1988 (53 FR 30140). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 31, 1988.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: November 18, 1987

Brief description of amendment: This amendment modified Section 6 of the Technical Specifications to reflect (I) a new Corporate and (II) a new plant staff organizational structure and (III) a revised composition of the Plant Operations Review Committee.

Date of issuance: October 31, 1988

Effective date: October 31, 1988

Amendment No.: 10

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

Date of initial notice in Federal Register: December 23, 1987 (52 FR 48589). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 31, 1988.

No significant hazards consideration comments received: No

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

Public Service Company of Colorado, Docket No. 50-267, Fort St. Vrain Nuclear Generating Station, Platteville, Colorado

Date of amendment request: December 2, 1986 as supplemented October 22, 1987 and July 15, 1988.

Brief description of amendment: The amendment modified paragraph 2.(D)(3) of the license to require compliance with the amended Physical Security Plan. The Plan was amended to conform to the requirements of 10 CFR 73.55. Consistent with the provisions of 10 CFR 73.55, search requirements must be implemented within 60 days and miscellaneous amendments within 180 days from the effective date of this amendment.

Date of issuance: October 24, 1988

Effective date: October 24, 1988

Amendment No.: 65

Facility Operating License No. DPR-34. Amendment revised the Technical license.

Date of initial notice in Federal Register: September 21, 1988 (53 FR 36673). The Commission's related evaluation of the amendment is contained in a Safeguards Evaluation Report dated October 24, 1988.

No significant hazards consideration comments received: No.

Local Public Document Room location. Greeley Public Library, City Complex Building, Greeley, Colorado

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

Date of application for amendment: March 7, 1988

Brief description of amendment: This amendment deleted license condition 2.C.(3) concerning relief from certain pump and valve testing requirements.

Date of issuance: October 26, 1988

Effective date: October 26, 1988

Amendment No.: 20

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

Date of initial notice in Federal Register: June 1, 1988 (53 FR 20045). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 26, 1988.

No significant hazards consideration comments received: No

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

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

Date of application for amendments: August 6, 1985 as supplemented on August 29, 1986 and August 16, 1988. The supplemental letters did not make technical changes to the original application.

Brief description of amendments: The amendments changed the Technical Specifications regarding air lock leakage testing.

Date of issuance: October 21, 1988

Effective date: October 21, 1988

Amendment Nos.: 89 and 62

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

Date of initial notice in Federal Register: September 25, 1985 (50 FR 38921). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 21, 1988.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: March 8, 1988 as supplemented August 31, and September 30, 1988.

Brief description of amendment: The amendment changes the Technical Specifications by revising Figures 3.9-1 and 3.9-2 of Section 3.9.12. These figures establish the minimum required fuel assembly exposure as a function of initial enrichment to permit storage of fuel assemblies in Regions 2 and 3 of the spent fuel assembly storage racks. In addition, the amendment revises Sections 5.3.1 and 5.6 of the Technical Specifications in terms of maximum initial enrichment of U-235 and minimum required burnup for Regions 2 and 3 of the spent fuel pool.

Date of issuance: October 28, 1988

Effective date: October 28, 1988

Amendment No.: 74

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

Date of initial notice in Federal Register: June 1, 1988 (53 FR 20046). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 28, 1988.

No significant hazards consideration comments received: No

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

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

Dates of application for amendment: May 20, 1988, as supplemented June 20, 1988, July 8, 1988, August 5, 1988, September 16, 1988, September 30, 1988, October 11, 1988, October 13, 1988, and October 24, 1988.

Brief description of amendment: The amendment changes the Technical Specifications to allow refueling and operating with (1) Vantage 5 (V5) improved fuel design in combination with the Westinghouse low parasitic fuel assemblies remaining in the core from Cycle 4 and (2) subsequent operating cycles with up to a full core of V5 fuel.

Date of issuance: October 28, 1988

Effective date: October 28, 1988

Amendment No.: 75

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

Date of initial notice in Federal Register: August 16, 1988 (53 FR 30144) The submittals dated August 5, 1988, September 16, 1988, September 30, 1988, October 11, 1988, October 13, 1988, and October 24, 1988 provided clarifying information that did not change the initial determination of no significant hazards consideration as published in the Federal Register. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 28, 1988.

No significant hazards consideration comments received: No

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

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: May 26, 1988

Brief description of amendment: The amendment revised Technical Specification Section 3.5.2, "Control Rod Insertion Limits," to assure reactor operation is consistent with core design analysis, in that the Control Group 1 (Shutdown Group) is precluded from insertion during power operation.

Date of issuance: October 21, 1988

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.: 111

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

Date of initial notice in Federal Register: July 27, 1988 (53 FR 28295). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 21, 1988.

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.

The Cleveland Electric Illuminating Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company, Docket No. 50-440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio

Date of application for amendment: June 9, 1988

Brief description of amendment: The amendment revises Table 3.8.4.1-1 of the Technical Specifications to delete spare

circuit breakers from the Table and correct typographical errors.

Date of issuance: October 24, 1988

Effective date: October 24, 1988

Amendment No.: 17.

Facility Operating License No. NPF-58. This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: September 15, 1988 (53 FR 35941). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 24, 1988.

No significant hazards consideration comments received: No

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, Ohio 44081

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

Date of application for amendment: June 28, 1988.

Brief description of amendment: The amendment changed the TS to reflect recent organizational changes. All references regarding the "Vice President, Nuclear" are changed to the "Senior Vice President, Nuclear." The position of General Manager, Engineering (Nuclear) has been deleted from the Nuclear Safety Review Board (NSRB) and the Manager, Licensing and Fuels, has been appointed Chairman of the NSRB.

Date of issuance: October 27, 1988.

Effective date: October 27, 1988.

Amendment No.: 39

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

Date of initial notice in Federal Register: August 24, 1988 (53 FR 32299). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 27, 1988.

No significant hazards consideration comments received: No.

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

Wolf Creek Nuclear Operating Corporation, Kansas Gas and Electric Company, Kansas City Power & Light Company, Kansas Electric Power Cooperative, Inc., Docket No. 50-482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: December 2, 1986 as supplemented February 18 and August 2, 1988.

Brief description of amendment: The amendment modified paragraph 2.E of the license to require compliance with the amended Physical Security Plan. This Plan was amended to conform to the requirements of 10 CFR 73.55. Consistent with the provisions of 10 CFR 73.55, search requirements must be implemented within 60 days and miscellaneous amendments within 180 days from the effective date of the amendment.

Date of Issuance: October 24, 1988

Effective date: October 24, 1988

Amendment No.: 21

Facility Operating License No. NPF-42. Amendment revised the license.

Date of initial notice in Federal Register: September 21, 1988 (53 FR 36674). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 24, 1988.

No significant hazards consideration comments received: No.

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

Yankee Atomic Electric Company, Docket No. 50-029, Yankee Nuclear Power Station, Franklin County, Massachusetts

Date of application for amendment: June 27, 1988

Brief description of amendment: The amendment changes the Technical Specifications to permit an increase in the nitrogen pressure in the safety injection accumulator.

Date of issuance: October 25, 1988

Effective date: October 25, 1988

Amendment No.: 119

Facility Operating License No. DPR-3: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: August 24, 1988 (53 FR 32304). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 25, 1988.

No significant hazards consideration comments received: No.

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

Yankee Atomic Electric Company, Docket no. 50-029, Yankee Nuclear Power Station, Franklin County, Massachusetts

Date of application for amendment: June 27, 1988

Brief description of amendment: The amendment revises the technical specifications to enable piping modifications needed to allow for installation of the Water Clean-Up System.

Date of issuance: November 1, 1988

Effective date: When the Water Clean-Up System is declared operable.

Amendment No.: 120

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

Date of initial notice in Federal Register: August 24, 1988 (53 FR 32304). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 1, 1988.

No significant hazards consideration comments received: No

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

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 December 16, 1988, 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).

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

Date of Application for amendment: October 10, 1988

Brief description of amendment: The amendment changed the Technical Specifications to allow an alternate sampling method of steam generator tube inspections, limited to the fourth refueling outage. Telephone authorization was granted on an emergency basis on October 14, 1988, and confirmed by letter dated October 14, 1988.

Date of Issuance: November 1, 1988

Effective Date: October 14, 1988

Amendment No.: 63

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

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

The Commission's related evaluation of the amendment, consultation with the State of New Jersey and final no significant hazards considerations determination are contained in a Safety Evaluation dated November 1, 1988.

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

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

NRC Project Director: Walter R. Butler

Dated at Rockville, Maryland, this 9th day of November, 1988.

For the Nuclear Regulatory Commission

Bruce A. Boger,

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

[Doc. 88-26331 Filed 11-15-88; 8:45 am]

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