123 Main Street
White Plains, New York 10601
914 681.6240

New York Power Authority

John C. Brons
Executive Vice President
Nuclear Generation

February 17, 1989
IPN-89-012

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

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Interim (180 day) Response to Generic Letter 88-14
"Instrument Air Supply System Problems Affecting Safety-Related Equipment"

Dear Sir:

As requested in the subject generic letter, the Authority has commenced a design and operations verification of the instrument air system for Indian Point Unit 3.

This program requires a substantial effort to complete. A description of the various tasks the Authority believes necessary to complete this effort is provided in Attachment 1.

The Authority has retained a contractor to perform the preliminary reviews and evaluations identified in the attachment. A substantial portion of the contractor's effort is complete and a preliminary report will be provided for Authority review shortly.

Indian Point 3 is currently shutdown for its Cycle 6/7 refueling and maintenance outage. Planning for this outage commenced and was to a large extent complete prior to the issuance of the subject generic letter. Instrument air system verification efforts requiring access to containment for inspection are expected to be completed during the Cycle 6/7 outage plan. Consistent with the staff position that instrument air system testing may have potentially adverse consequences during power operation and, therefore, may be deferred until the next scheduled outage, the Authority plans to complete the requested verification, including any testing determined necessary, prior to startup from the next (Cycle 7/8) refueling outage.

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Operators verify and log system parameters such as pressure, temperature and filter delta-p each shift. Dew point measurements are taken periodically to minimize water accumulation and the potential for freeze-ups. The system employs oil-free compressors (i.e. no oil required for piston/cylinder lubrication). Nevertheless, the system is provided with several stages of oil filtration/separation equipment to minimize oil/oil vapor carryover. This equipment is monitored and maintained. The system is equipped with particulate filters. These are serviced based on elevated delta-p across the filter.

A preventive maintenance program is in place for the air compressors. Maintenance procedures are in place for the instrument air dryers and for compressor lubrication. Operations and maintenance personnel receive training on compressor operation and maintenance applicable to their areas of responsibility.

An alarm response procedure (ARP-12, "Alarm Response Procedure Panel SJF-Cooling Water and Air") is used to respond to alarms generated by the instrument air system. An off-normal operating procedure (ONOP-IA-1, "Loss of Instrument Air") is used for recovery in the event of a complete loss of instrument air. Normal operator classroom training is augmented with loss of instrument air scenarios on the simulator.

The Authority anticipates no major system modifications as a result of the efforts it is undertaking in response to the subject generic letter. Only minor enhancements to procedures and practices are expected in order to establish the confidence level desired by the staff.

Should you or your staff have further questions regarding this letter, please contact Mr. P. Kokolakis of my staff.

Very truly yours,

John C. Brons

John C. Brons
Executive Vice President
Nuclear Generation

STATE OF NEW YORK
COUNTY OF WESTCHESTER

Subscribed and sworn to before me this
17th day of February, 1989

Mina Holden

Notary Public

MINA HOLDEN
NOTARY PUBLIC, State of New York
Westchester County
No. 4829150
My Commission Expires Aug. 31, 1989

Attachment

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

Resident Inspector's Office
Indian Point Unit 3
U.S. Nuclear Regulatory Commission
P.O. Box 337
Buchanan, NY 10511

Joseph D. Neighbors, Sr. Proj. Mgr.
Project Directorate I-1
Division of Reactor Projects I/II
U.S. Nuclear Regulatory Commission
Mail Stop 14B2
Washington, D.C. 20555

ATTACHMENT 1
G/L 88-14 SUBTASKS

1. Review and evaluate Maintenance Machine History files and other documents related to the instrument air system, such as but not limited to the following: Maintenance Work Requests (MWR's), Licensee Event Reports (LER's), Non-Conformance Report (DCAR's).

2. Evaluate the instrument air system design bases by reviewing system descriptions, piping and instrument drawings, design criteria, equipment procurement specifications, vendor manuals, and safety analysis report. This review will be used to identify instrument air related safety functions, performance requirements and instrument air quality requirements for the individual components utilizing instrument air, and additionally identify manufacturer's requirements for the system components.

3. Evaluate instrument air interfaces such as accumulators, dryers and other components which provide a safety function or are used to maintain air quality.

4. Review and evaluate air system testing records, initial acceptance testing and periodic air quality surveillance testing to ensure testing adequately measured the response of the system to both sudden loss of pressure and gradual loss of pressure.

5. Review and assess the preventive maintenance practices and procedures related to the instrument air system.

6. Review and evaluate normal and emergency operating procedures.

7. Evaluate training on the instrument air system and determine its adequacy.

8. Conduct system walkdowns to assure design specifications and appropriate manufacturer's requirements have been met.

9. Verify/test instrument air quality and compare to component specifications and manufacturer's requirements.

10. Conduct failure analysis on instrument air system components to identify single point failures and common mode failures that have the potential of compromising interfacing safety systems.

NewYorkPower Authority

John C. Brons
Executive Vice President
Nuclear Generation

February 17, 1989
IPN-89-012

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
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Very truly yours,

John C. Brons

John C. Brons
Executive Vice President
Nuclear Generation

STATE OF NEW YORK
COUNTY OF WESTCHESTER

Subscribed and sworn to before me this
17th day of February, 1989

Mina Holden

Notary Public

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NOTARY PUBLIC, State of New York
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cc: U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Resident Inspector's Office
Indian Point Unit 3
U.S. Nuclear Regulatory Commission
P.O. Box 337
Buchanan, NY 10511

Joseph D. Neighbors, Sr. Proj. Mgr.
Project Directorate I-1
Division of Reactor Projects I/II
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Docket file w/encl
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JNeighbors
JScinto

February 9, 1989

DOCKET NO(S). 50-286

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

SUBJECT: POWER AUTHROITY OF THE STATE OF NEW YORK
Indian Point Nuclear Generating Unit No. 3

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

- [ ] Notice of Receipt of Application, dated ____________.
- [ ] Draft/Final Environmental Statement, dated __________.
- [ ] Notice of Availability of Draft/Final Environmental Statement, dated ____________.
- [ ] Safety Evaluation Report, or Supplement No.____________________ dated __________.
- [ ] Environmental Assessment and Finding of No Significant Impact, dated ____________.
- [ ] Notice of Consideration of Issuance of Facility Operating License or Amendment to Facility Operating License, dated __________.
- [X] Bi-Weekly Notice; Applications and Amendments to Operating Licenses Involving No Significant Hazards Considerations, dated 2/1/89 ~~xxxxxxxxxxxx~~ comments by 3/3/89 .
- [ ] Exemption, dated __________.
- [ ] Construction Permit No. CPPR-__________, Amendment No. ____________ dated __________.
- [ ] Facility Operating License No. ___________, Amendment No. ___________ dated __________.
- [ ] Order Extending Construction Completion Date, dated __________.
- [ ] Monthly Operating Report for ________________ transmitted by letter dated __________.
- [ ] Annual/Semi-Annual Report- ____
  ________________________ transmitted by letter dated __________.

Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: See next page

M/A 4

CC

OFFICE	PDI-1						
SURNAME	CVogan:VR						
DATE	2/13/89						

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statements should notify the ACRS Staff member named below as far in advance as practicable so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC Staff, its consultants, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the cognizant ACRS Staff member, Mr. Sam Duraiswamy (telephone 301/492-9522) between 7:30 a.m. and 4:15 p.m. Persons planning to attend this meeting are urged to contact the above named individual one or two days before the scheduled meeting to be advised of any changes in schedule, etc., which may have occurred.

Date: January 25, 1989.

Morton W. Libarkin,

Assistant Executive Director for Project Review.

[FR Doc. 89-2299 Filed 1-31-89; 8:45 am]

BILLING CODE 7590-01-M

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Subcommittee on Safety and Research Program; Meeting

The ACRS Subcommittee on Safety Research Program will hold a meeting on February 8, 1989, Room P-114, 7920 Norfolk Avenue, Bethesda, MD.

The entire meeting will be open to public attendance.

The agenda for subject meeting shall be as follows: *Wednesday, February 8, 1989—8:30 a.m. until 1:00 p.m.*

The Subcommittee will discuss the ongoing and proposed NRC Safety Research Program and budget.

Oral statements may be presented by members of the public with concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral

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 December 30, 1988 through January 12, 1989. The last biweekly notice was published on January 11, 1989 (54 FR 1018).

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION AND OPPORTUNITY FOR HEARING

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

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination. The Commission will not normally make a final determination unless it receives a request for a hearing.

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

By March 3, 1989, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written petition for leave to intervene. Requests for a hearing and petitions for leave to intervene shall be filed in accordance consequences of previously analyzed accidents.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated because this change places administrative controls on fuel oil inventory. No unanalyzed accidents can result from this change. No change in plant system's configuration is required, therefore, no new accident or different kind of accident than previously evaluated can be introduced.

3. Operation of the facility, in accordance with the proposed amendment, would not involve a significant reduction in a margin of safety because this increase in minimum diesel fuel inventory improves margins of safety by assuring seven days operation of one diesel at its rated design capacity. Additional fuel oil inventory adds to the margin of safety. This results from providing sufficient fuel oil to allow the diesels to be operated to their rated design capacity without limiting operation to only minimum required safety features.

The staff has reviewed CP&L's evaluation and agrees with its analysis. Accordingly, the Commission proposes to determine that the proposed change does not involve a significant hazards consideration.

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

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. 50-373 and 50-374, LaSalle County Station, Units Nos. 1 and 2, LaSalle, 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 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 organization 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. Furthermore, 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 "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: Public Library of Illinois Valley Community College, Rural Route No. 1, Ogelsby, Illinois 61348.

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

NRC Project Director: Daniel R. Muller

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

Date of amendment request: September 23, 1988

Description of amendment request: The proposed amendment would change the Technical Specifications (TS) and associated Bases dealing with "Reactor Coolant System Leakage and Leakage into the Containment Free Volume," as follows:

(1) TS 3.1.F and the associated Basis would receive minor editorial changes including the repagination of text for increased spatial uniformity, the correction of minor typographical errors.

(2) TS 3.1.F.1.b.(6) would be modified to require grab sample analysis upon inoperability of either (rather than both) of the radioactivity monitoring systems required by TS 3.1.F.1.a.(6).

(3) The TS 3.1.F Basis, TS page 3.1.F-4, paragraph a, would be revised to state that the containment air particulate monitor would meet "the recommended sensitivity guidelines of Regulatory Guide 1.45." Paragraph "b" would be revised to remove the activity levels stated (1E-2 uc/cc and 1E-7 uc/cc).

(4) The associated Technical Specification 3.1.F Basis would remove statements that the containment air particulate monitor is sensitive to 0.025 gpm to greater than 10 gpm so that an increase in reactor coolant system leakage of 1 gpm would be detectable

Technical Specifications. Therefore, it does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not change the design or function of any equipment and has no impact on any accident analyses. The changes are being made for consistency only and can clearly be classified as administrative. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not impact any safety analyses because it is purely administrative in nature. No modification or change to equipment is involved. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

The staff has reviewed the licensee's determination and is in agreement with them. Accordingly, the Commission proposes to determine that these changes do not involve a significant hazards consideration.

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

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

NRC Project Director: Elinor G. Adensam

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

Date of application for amendments: August 23, 1988

Description of amendments: The proposed amendment would revise Technical Specification Table 3.3.5.7-1 to change the type of fire detection instruments required for the diesel generator building.

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 (CP&L or the licensee) provided the following analysis to support the finding that the proposed changes do not involve a significant hazards consideration:

1. The fire detection system is designed for detection and mitigation of accidents involving fires but does not affect the probability of those accidents. The proposed change involves the use of a more effective fire detection instrument arrangement based on the fire hazards associated with the diesel generator cells. The new heat and flame detectors will provide an enhanced means of detecting the major fire hazard (a fuel oil fire) expected to result in significant damage to the diesel generators. This enhancement will provide equivalent limitations on the consequences of this type of fire. 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. The accidents analyzed in Chapter 15 of the Updated FSAR are not affected by the proposed change in detector types. The change in detectors will not affect the fire detection system's ability to perform its intended function. The new flame and heat detectors are more suited for the fire hazard involved than the present smoke detectors. Since the fire detection system, including the detection instruments, affect [SIC] only the detection and subsequent mitigation of a fire, 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 proposed change provides an enhanced capability of detecting a severe fuel oil fire in the diesel generator cells. This type of fire hazard was evaluated to have the greatest impact on the continued availability of the diesel generators. The new flame and heat detectors are more suited for the rapid detection of this type of fire. The present smoke detectors are better suited for detecting a fire with a long incipient or smoldering stage, which generally involves burning of ordinary combustible materials. It has been determined that this type of fire will not likely result in damage to safety related equipment located in the diesel generator cells. Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

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 analysis and agrees with the determination. 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, Docket No. 50-261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: November 30, 1988

Description of amendment request: The request would amend the Technical Specifications (TS) to increase the minimum inventory of diesel generator fuel at the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBR-2), site. The existing diesel fuel storage capacity is sufficient to meet the requirements of the proposed TS amendment. The proposed TS would correct an inconsistency between the TS and the HBR-2 updated Final Safety Analysis Report.

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 (CP&L) has reviewed the Technical Specifications change request in accordance with the standards set forth in 10 CFR 50.92 and concluded that this change does not constitute a significant hazards consideration based upon the following:

1. Operation of the facility, in accordance with the proposed amendment, would not involve a significant increase in the probability or consequences of an accident previously analyzed because increasing the required fuel oil inventory increases the length of time the diesels can function before resupply is necessary. The change to administratively maintain an increased minimum fuel oil inventory in the Unit 1 tank does not impact the combustible loading for the Unit 2 Fire Hazard Analysis. Since no change in plant system's configuration is required to achieve the inventory increases, nor does any fuel oil storage system contribute to any previously analyzed accident sequence, the proposed change cannot increase the probability or

background setpoint for containment purge and exhaust isolation.

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 Technical Specifications list the operating range for the four containment radiation monitors for isolation of purge and exhaust in the unlikely event of an accident inside containment. These ranges were derived from the Combustion Engineering Standard Technical Specifications and from Generic Letter 82-16 which used counts per minute. The original numbers were proposed for the monitors and the licensee has determined that at the lower end of the range requirements (per the Technical Specification), the monitors are not accurate or reliable and would be called inoperable. This places the plant in an unwarranted position in some modes and penalizes operation. The setpoint is set at twice background for isolation of purge and exhaust but background at the low end of the range may be below the monitor capability to measure. The monitor reliability and accuracy begins at 20 mR/hr and goes up to $5x10^6$ mR/hr. The licensee proposes to incorporate this accurate range in the Technical Specifications and to isolate purge and exhaust at readings twice the lowest accurate reading or twice background whichever is the highest. Exceeding the setpoint will initiate the logic for the Containment Purge Isolation Signal (CPIS). CPIS is also initiated by monitors in the exhaust stack which is the referenced operation in the Off-Site Dose Calculation Manual (ODCM). The exhaust stack monitors and the ODCM are not changed by the proposed amendment.

The licensee has provided an analysis of the hazards consideration of the proposed change.

CPIS guards against fuel handling accidents because the inside-the-containment accident ranks highest in off-site dose consequences, excepting LOCAs. Large radioactive releases into containment from LOCAs cause a Safety Injection Actuation Signal (SIAS) and Containment Isolation Actuation Signal (CIAS) to protect the public. The maximum differential pressure expected during accident conditions dictates the minimum time to isolate the purge butterfly-valves. Raising the operating range of the containment area radiation monitors... does not change the ability of the area radiation monitors (i.e., CPIS) to protect against fuel handling accidents in containment. Further, the change does not affect the plant stack monitors which also cause CPIS. The probability and consequences of a fuel handling accident remain as currently analyzed.

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

The containment radiation area monitoring machines do not change because of this request. The ability for them to measure the operating range remains unchanged. The alarm/trip setpoint still happens at twice the background level the machine can detect (i.e., 40 mR/h or higher). The analysis of the fuel handling accident without containment isolation meets 10 CFR 100 criteria while using an assumed containment air dose rate much larger than the new 20 mR/h low range number. The new setpoint does not affect the SIAS and CIAS logics protecting the public during a LOCA. The area radiation monitor machines remain unchanged by this request so no new or different failure modes exist.

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

The Operability of the radiation monitoring channels ensure that: (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident.

The new operating range for the containment area monitors retains the bases of its specification. The area radiation monitors measure radiation levels in their locations within the capability of the machines. The monitors still "trip" upon reaching the licensee determined setpoint. Action Statements for Table 3.3-6 ensure sufficient radiation field information exists after an accident. Regulatory Guide 1.97 requires the lowest containment area radiation monitor reading to be 0.1 R/h (100 mR/h). Therefore, the new operability range provides personnel protection no different than before because Waterford used the same machines, plus fulfilling regulatory requirements for containment area radiation monitoring.

Therefore, the proposed change will not 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: 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

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

Date of application for amendment: November 22, 1988

Description of amendment request: This amendment request would amend sections of the Maine Yankee Atomic Power Station Technical Specifications by extending the testing intervals of the Main Steam Excess Flow Check Valves and Turbine Valves. Since both of these tests involve power reductions, Maine Yankee Atomic Power Company (the licensee) is proposing to perform these tests at the same interval to minimize power reductions.

First, the proposed amendment would modify Technical Specification 4.6, "Periodic Testing" by extending the specified surveillance testing interval for Main Steam Excess Flow Check Valves from once every six weeks to once every three months.

Main Steam Excess Flow Check Valves (EFCVs), designed to limit an excessive reactor coolant system cooldown rate and resultant reactivity insertion following a main steam break incident, are tested to ensure that the valves are not mechanically prevented from closing when needed.

Second, the proposed amendment would modify Technical Specification 4.2, "Equipment and Sampling Tests," by extending the surveillance requirements for turbine stop, governor, reheater and intercept valves, from once per month to once per three months, consistent with the proposed interval for the EFCVs.

Turbine valves are periodically tested to ensure they are operable and are designed to protect the turbine from excessive overspeed. An evaluation of turbine overspeed and missile generation probability considering turbine valve test interval has been performed by the licensee.

Basis for proposed no significant hazards consideration: The proposed changes to the Technical Specifications have been evaluated by the licensee to determine whether they constitute a significant hazards consideration as required by 10 CFR Part 50, Section 50.91

within one minute after it occurs and the containment radiogas monitor is less sensitive than the air particulate monitor.

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.

The licensee provided the following analysis of the proposed changes:

In accordance with the requirements of 10 CFR 50.92, the proposed Technical Specification change is deemed to involve no significant hazards considerations because operation of Indian Point Unit No. 2 in accordance with this change would not:

1. Involve a Significant Increase In The Probability of Consequences of An Accident Previously Evaluated:

The proposed changes to TS 3.1.F (Reactor Coolant System Leakage and Leakage into Containment Free Volume) merely make the Indian Point Unit 2 leakage detection system requirements for the containment radioactivity monitors consistent with those of Regulatory Guide 1.45.

This Regulatory Guide has been endorsed by the staff on a generic basis, and it is now applied to IP-2. The analysis supporting the application of Regulatory Guide 1.45 to IP-2 has been provided to the staff in the [Leak Before Break] LBB application submittal on May 23, 1988. It is apparent, therefore, that this change does not involve a significant increase in the probability or the consequences of an accident previously evaluated, but applies an accepted standard (Regulatory Guide 1.45) to substantiate the basis of this TS. This TS as written is unnecessarily restrictive and does not provide any additional safety benefits. The proposed change to the LCO action statement, 3.1.F.1.b.(6), imposes additional requirements on containment radioactivity monitors and does not increase the probability or consequences of an accident previously evaluated.

The additional changes requested are purely administrative in nature and only change typographical errors, make minor editorial changes for consistency, repaginate the document and delete pertinent portions of the IP-2 Technical Specification that are no longer effective or have been previously approved for deletion, therefore these changes do not increase 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:

The request to amend the TS merely allows conformance to Regulatory Guide 1.45 and imposes more restrictive requirements on the containment radioactivity monitors; it does not affect the reliability or the adequacy of the leakage detection system currently at IP-2, nor does it affect the design basis as described in the FSAR, in and of itself. The related submittal, which requests authorization for the use of the LBB methodology as specified in the final rule, also does not alter the existing plant design, but merely eliminates the necessity of considering the dynamic effects of Double Ended Guillotine Break (DEGB) on the primary system. Neither [these] TS changes nor the basis for these changes (LBB/Regulatory Guide 1.45) creates the possibility of a new or different kind of accident, because the design itself is not being changed by this TS. The TS is merely being updated to include current accepted standards.

The additional changes requested are purely administrative in nature and only change typographical errors, make minor editorial changes for consistency, repaginate the document and delete pertinent portions of the IP-2 Technical Specification that are no longer effective or have been previously approved for deletion, therefore these changes do not create the possibility of a new or different kind of accident.

3. Involve a Significant Reduction in Margin of Safety:

Revising the LCO action and basis for the radioactivity monitors has a negligible effect on the margins of safety. The purpose of the containment air particulate monitors is to detect sufficiently small amounts of reactor coolant leakage to indicate an unacceptable plant condition, e.g., pipe cracks or excessive valve or seal leakage. The accepted level of sensitivity sought by the staff is as set forth in Regulatory Guide 1.45, and this TS change will allow IP-2 to conform to this accepted guidance. The IP-2 TS basis as set forth at present is based on optimum instrument function, not safety requirements, and thus is far too restrictive. Changing the TS to conform with Regulatory Guide 1.45 will not involve a significant reduction in the margin of safety.

The additional changes requested are purely administrative in nature and only change typographical errors, make minor editorial changes for consistency, repaginate the document and delete pertinent portions of the IP-2 Technical Specification that are no longer effective or have been previously approved for deletion, therefore these changes do not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Robert A. Capra, Director

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

Date of amendment request: December 23, 1988

Description of amendment request: The proposed amendment would change the Technical Specifications to correct the terminology of control room isolation for toxic gas protection action.

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 "(i). A purely administrative change to technical specifications: for example... correction of an error...". The proposed change is directly related to this example in that the Technical Specifications currently reference a recirculation mode for the toxic gas event and recirculation means pressurizing the control room with outside air. In the toxic gas event, outside air is not permitted in the control room and the proper reference should be isolation. In making the change to the Technical Specification, nothing will change and only the Technical Specification word is corrected.

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

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

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

NRC Project Director: Jose A. Calvo

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

Date of amendment request: December 23, 1988

Description of amendment request: The proposed amendment would change the Technical Specifications to incorporate the correct operating range for the Containment Area Radiation Monitors and clarify the radiation

(5) Loss of Feedwater
(6) Loss of Coolant Flow
(7) Steam Line Rupture
(8) Large Break LOCA
(9) CEA Drop
(10) Loss of Load
(11) CEA Ejection

Other transients that required a partial reanalysis of review included:
(1) Steam Generator Tube Rupture
(2) Small Break LOCA
(3) Containment Overpressure

In each case the reanalysis demonstrated that the applicable acceptance criteria for the accident or transient continue to be met. For the transients requiring a partial reanalysis or review, the parameters were bounded by previously approved safety analyses and therefore are not adversely affected by the power upgrade.

In summary, our evaluation of accidents previously analyzed in the FSAR has demonstrated that all applicable acceptance criteria continue to be met. Therefore, the proposed Technical Specification changes for Cycle 11 operation at 2700 MWth do not significantly increase the consequences of an accident previously evaluated.

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

The core for Cycle 11 operation at 2700 MWth is similar in fuel design, CEA placement, thermal, thermal-hydraulic, and physics characteristics to that of both Cycle 10 and Cycle 11 at 2630 MWth operation. We have concluded that Cycle 11 operation at 2700 MWth does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The design of Cycle 11 at 2700 MWth is similar to both Cycle 10 and Cycle 11 at 2630 MWth operation. The methods used to analyze Cycle 11 operation at 2700 MWth are the same as were used for both Cycle 10 and Cycle 11 at 2630 MWth operation which have been previously approved by the NRC staff. Additionally, the safety analysis acceptance criteria for operation at 2700 MWth during Cycle 11 have not changed from that used in previous reload submittals. We have demonstrated that these acceptance criteria continue to be met. We have therefore concluded that Cycle 11 operation at a rated power level of 2700 MWth does not involve any significant reduction in a margin of safety.

Based on the reasons discussed above, the licensee has concluded that the proposed changes to Technical Specifications do not involve a significant hazards consideration as defined by 10 CFR 50.92. The Commission agrees with this conclusion.

Local Public Document Room location: Wiscasset Public Library, High Street, P. O. Box 387, Wiscasset, Maine 04578.

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

Date of amendment request: September 23, 1988, as supplemented November 30, December 16 and December 21, 1988.

Description of amendment request: The amendment would change the Technical Specifications (TS) by adding a plant service water radiation monitor in TS 3/4.3.7.1, "Radiation Monitoring Instrumentation," and adding two valves in TS 3/4.8.4.2, "Motor Operated Valves Thermal Overload Protection." These new TS are proposed for an alternate decay heat removal system (ADHRS) to be installed for use during refueling outages when the residual heat removal (RHR) system is out of service for maintenance or inspection. The proposed ADHRS would use plant service water (PSW) in the ADHRS heat exchangers to remove decay heat from reactor cooling water. The service water radiation monitor would detect heat exchanger tube leakage. The ADHRS would consist of two pumps and two heat exchangers in parallel with a common suction line connected to the existing RHR suction line and a common discharge line connected to one of the low pressure coolant injection (LPCI) lines. The two valves added to TS 3/4.8.4.2 would be used to isolate the ADHRS from the RHR and the LPCI lines during plant operation. The ADHRS would not be a safety related system.

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's analysis of the proposed amendment against the three standards in 10 CFR 50.92 is reproduced below.

I. These changes would not significantly increase the probability or consequences of an accident previously evaluated.

(a) The ADHRS (including operating requirements) has been found not to significantly increase the probability of previously evaluated accidents and events as discussed below.

The nature of the ADHRS and its interconnections with existing plant systems results in specific consideration of the following events for possible increases in probability:

o Pipe break or leak
o Loss of coolant accident (LOCA)/inadvertent reactor vessel drainage
o Internally generated missiles
o Reactor coolant pressure boundary (RCPB) overpressure
o Fire
o Loss of electrical power
o Offsite radioactive release

Other events/accidents are excluded from detailed consideration because there are not identified functional (physical connections or controls interface) relationships and because the ADHRS has been designed to preclude any adverse spatial or environmental interactions. Examples of such events include secondary plant transients, loss of instrument air, reactivity control failures, and fuel handling accidents.

Pipe Break or Leak

The ADHRS will not significantly increase the probability of a pipe break or leak in existing systems, and the probability of failure of the ADHRS fluid components is no greater than existing systems, as presently analysed [SIC] in UFSAR Section 3.6.

The design of the ADHRS is such that stress loadings due to the attachment of ADHRS piping equipment to existing plant fluid systems does not cause applicable stress allowables on existing components to be exceeded. This is documented by calculations that employed methodologies, assumptions, loading combinations, and other criteria the same or equivalent to those contained in UFSAR Section 3.9.

Evaluated were mechanical loadings imposed by the ADHRS, changes in the design pressure and temperature of existing piping, and fatigue considerations. Attachment loadings were evaluated at the suction and discharge connections of the ADHRS to the RHR and PSW systems, at valves E12F066A and B due to the addition of motor operators to these valves, loads imposed by the new valve added to the vertical leg of the RHR suction line from the spent fuel pool, and at locations where lead wrap pipe shielding was added. Existing piping and supports were verified to be adequate or supports are to be modified and/or new supports added as applicable to achieve acceptable conditions.

The added ADHRS piping handling reactor coolant is designated as safety-related, ASME Section III, Class 2 and Class 3, and Seismic Category I. In addition, PSW piping inside the RHR "C" pump room and up to the isolation valves at the existing supply and return headers and air handling unit is designated as safety-related, ASME Section III, Class 3, and Seismic Category I. Other piping is B31.1 and designed for SSE loads. The design and analysis of this piping is to the same standards, criteria, and methodology as existing plant piping. Damage to the ADHRS piping and components due to over-pressure is prevented by design of the system for the highest pressure possible from ADHRS operation (pump shutoff head plus maximum static head at the pump suction), and providing for system isolation from higher pressure sources (i.e., RHR "C" pump discharge). The portion of the system that cannot be isolated from RHR "C" (downstream of the ADHRS isolation valves) is designed for RHR "C" design pressure. Thermal relief valves are provided in the ADHRS for overpressure protection when the system is isolated and for minor leakage that may occur across the boundary valves. Adequate vents and drains are included in the design to provide for complete filling and venting prior to pump start. This provision will preclude the possibility of a waterhammer event during startup of the system.

Materials (mostly carbon steel) used in the ADHRS pressure boundary are compatible with those used in connected systems and will not induce a degradation of existing piping. The ADHRS does not employ any stainless steel pressure boundary components other than instrument connections. Reactor coolant chemistry is controlled to prevent failure of stainless steel components due to intergranular stress corrosion cracking. On the reactor coolant side of the ADHRS, no features are provided that would alter reactor coolant chemistry.

Erosion of existing and new piping is not expected to occur based on flow velocities and expected usage times. Specified nominal corrosion allowances for new piping are the same as for existing piping, and excessive corrosion during lengthy inactive periods is to be precluded by wet layup of the reactor water side (the same as currently performed for the existing RHR system), and layup of the PSW side.

Adequate protection from overpressurization of piping is provided by the inclusion of appropriate isolation capability at system interfaces.

Consequently, the ADHRS modification does not significantly increase the probability of a break in existing piping and the probability of a break in the new piping is no higher than in existing piping.

LOCA/Inadvertent Vessel Drainage

Such events as a consequence of the presence or use of the ADHRS can only occur during operational conditions 4 or 5 (cold shutdown or refueling), since in all other operational conditions the ADHRS is required to be physically isolated from other systems.

The probability of a pipe break outside containment is not significantly increased by the addition of the ADHRS. Per UFSAR Section 3.6, postulating a pipe failure when the ADHRS is aligned or operating is not required. Specifically, UFSAR Section 3.6A.1.1c states that "Pipe breaks or cracks were postulated to occur during normal plant operation (i.e., reactor startup, operation at power, hot standby or reactor cooldown to cold shutdown)." These conditions do not apply to operational conditions 4 or 5 when the ADHRS would be in operation.

To the extent that the criteria cited in Section 3.6 implicitly assume adequate piping design, the ADHRS satisfies this assumption. As for a pipe break inside containment, the only piping involved is the RHR "C" LPCI injection line to the reactor vessel. This line was evaluated for an increased fatigue usage factor arising from a longer operating time for this pipe (albeit at a lower flow rate) due to ADHRS operation and determined to remain within code limits. Therefore, addition of the ADHRS will not significantly increase the probability of failure of this pipe.

Relative to a LOCA caused by inadvertent drainage of the reactor vessel (i.e., a system alignment that allows either gravity or pumped flow from the vessel via an existing isolation point), the ADHRS design and its accompanying procedural requirements make this a no more probable event than that associated with existing plant systems.

The probability of a LOCA type event is not significantly increased by the addition of the ADHRS.

Internally Generated Missiles

The ADHRS does not pose any additional hazards in this regard as a source of missiles. The ADHRS is not postulated to be a source of pressurized component missiles, and potential rotating component missiles (from the pumps and air handling unit fan) are also not required to be postulated.

Reactor Coolant Pressure Boundary Overpressure

When in operation, the ADHRS pumps reactor coolant to the reactor vessel. The only time when ADHRS could be operated that reactor pressure could potentially increase is in operational condition 4 with the reactor vessel head installed. However, the maximum discharge pressure of the ADHRS is well below reactor vessel design pressure. The ADHRS otherwise will not interfere with vessel overpressure protection functions so that the probability of an event of this kind is not increased.

Fire

The ADHRS does not involve the addition of any combustible loading in excess of that assumed in the FHA to the areas in which it is located as contained in the GGNS Fire Hazards Analysis Report and thus does not contribute to the increased probability of a fire.

Loss of Electrical Power

The only Class 1E electrical loads involved with the ADHRS design are the motor operators for valves E12F066A and B. These power supplies are designed in accordance with the various Class 1E criteria described in UFSAR Section 8. The power supplies to these motors are adequate for the additional loading. Thus, there is no additional potential for loss of Class 1E power. The added non-Class 1E loads have been designed to standards consistent with the existing design for the type of load. Therefore, to the extent that loss of non-Class 1E power might initiate a plant transient, no greater probability is indicated.

Offsite Radioactive Release

The ADHRS design provides a boundary, within the ADHRS heat exchangers, between radioactively contaminated reactor coolant and the PSW system which can discharge to the environment. The boundary (heat exchanger tubes) is designed and constructed to ASME Section III, Class 3 standards and to Seismic Category I criteria. As such the probability of its failure is no greater than other plant systems containing radioactive fluids.

Inadvertent Operation of RHR Shutdown Cooling

This event is discussed in UFSAR Section 15.1.6 and relates to a slow decrease in moderator temperature leading to an increase in reactor power, when the reactor is critical or near critical. The only cause for this event cited in the UFSAR is misoperation of the cooling water controls for the RHR heat exchanger.

Inasmuch as the cooling water controls for the ADHRS are no more prone to failure than those of the RHR (both are manually operated with no automatic activation or automatic control), and there would be no reason for greater operator error, the ADHRS would not significantly increase the probability of this event.

(b) The ADHRS design (including operating requirements) does not significantly increase the consequences of an accident previously evaluated in the UFSAR as discussed below.

The assessment relative to consequences results in specific consideration of the following events for significant increases in consequences.

o External high wind/tornado, tornado missile, and flooding events.
o Seismic events
o Pipe break or leak
o Internally generated missiles
o LOCA
o Hydrodynamic events
o Loss of electrical power or instrument air
o Fire
o Offsite radioactive release

Other events/accidents are excluded from detailed consideration since there are no identified functional interactions or spatial/environmental interactions.

External high wind/tornado missile and flooding

Components added and employed by the ADHRS are completely housed within existing safety-related portions of the plant (Auxiliary Building) and as such are not subject to damage or other consequences of these extreme environmental effects which are discussed in UFSAR Sections 3.3, 3.4 and 3.5.

Seismic Events

The ADHRS and its components (including existing plant equipment used for the ADHRS) are designed or analyzed for safe shutdown earthquake (SSE) design basis loads in order to preserve the pressure integrity of safety-related components, the operability of the motor operators and conduit for valves E12F066A and B, and to ensure that non-safety related components of the ADHRS do not pose a hazard to safety-related plant components during a seismic event. Existing components connected to the ADHRS have been re-evaluated for the new attachment loadings. This re-evaluation also includes assurance of continued operability of any affected active safety-related components (e.g., motor operated valves) during or after the seismic event as may be required.

The structural integrity of the Auxiliary Building due to loads imposed by the ADHRS is adequate. The major ADHRS components are located on the building basement at El. 93', and there is no appreciable effect on structural seismic response. Loads imposed on walls, beams, or other structural elements in the ADHRS design are acceptable.

Pipe Break/Leak

The ADHRS does not present hazards (pipe whip, jet impingement, etc.) from high energy line breaks (HELB) and is not subject to damage from HELBs in other plant systems.

The ADHRS safety functions are not impacted by spray or flooding from moderate energy pipe cracks in other plant systems (no such events are required to be postulated during operational conditions 4 and 5 when the ADHRS is in operation). The ADHRS also does not pose a hazard to other safety-related plant components due to spray effects of a pipe crack. Existing drainage and detection capabilities are adequate for any flooding that may be involved in the event of an ADHRS line crack or break.

Therefore, the addition of the ADHRS does not significantly increase the consequences of pipe break as discussed in UFSAR Section 3.6.

Internally Generated Missiles

The potential for damage to the ADHRS, and therefore increased accident consequences due to internally generated missiles (valve stems, rotating equipment, etc.) or turbine missiles has been assessed in accordance with UFSAR Sections 3.5.1.3 and 3.5.1.4. No additional consequences were determined to occur.

LOCA

Postulations concerning a LOCA in reactor operational conditions 4 and 5 are not contained in the UFSAR, and such an event is not analyzed (UFSAR Sections 6.2 and 15.6 deal with the design basis LOCA occurring during reactor operation).

However, to the extent that the ADHRS could either interfere with the mitigation of such an event or be the cause of the event, no increased consequences have been identified.

The controls interaction and system interaction evaluation completed for the ADHRS indicate that no interference by the ADHRS in emergency core cooling system (ECCS) functions or vessel level detection functions would occur. Protection against inadvertent drainage during operation of ADHRS is equivalent to that provided for existing plant systems through the use of interlocks and restricting operating procedures. Additionally, a failure of the ADHRS (i.e., a moderate energy pipe crack) is not required to be postulated during times when it is conveying reactor coolant (UFSAR Section 3.6A.1.1.C).

On the above basis, the ADHRS modification would not significantly increase the consequences of a LOCA.

Hydrodynamic Load Events

Such events include LOCAs (UFSAR Section 15.6), inadvertent SRV opening (UFSAR 15.1.4), and safety-relief valve operation incident to other transients or accidents. While the ADHRS would not be in operation during such events (as there is no reactor pressure source in operational conditions 4 or 5), it is nevertheless designed for such loads. Also connecting equipment was reanalyzed to the extend that these loads may be transmitted to the ADHRS, to ensure that the ADHRS would not be a hazard and thereby increase the consequences of such events.

Loss of Electrical Power or Instrument Air

The ADHRS has no safety-related function relevant to the loss of AC power as discussed in UFSAR Section 15.2.6 or instrument air as discussed in UFSAR Section 15.2.10 and no reliance on electrical power or instrument air is made by ADHRS for other plant events or accidents to ensure that consequences are not increased beyond those already evaluated.

Fire

The existing fire detection and suppression equipment is adequate for any potential fire originating within the ADHRS or incident to its presence.

Other than the motor-operators for valves E12F066A and B, and associated power and control cable, the interlock for valve E12F004C, and the pump start permissive bypass switches for valves E12F066A and B no other safety-related components are susceptible to damage by fire. To the extent that the functional capability of the E12F066A or B operators may be lost due to a fire, no increase in accident consequences is indicated. This is because the events for which they are provided (e.g., LOCAs) are not postulated to occur coincident with a fire, as indicated in the GGNS Fire Hazards Analysis Report and 10 CFR Part 50 Appendix R. The isolation function of the valves is only required during operational conditions 4 and 5. As such the ability to bring the plant to a cold shutdown condition in the event of a fire is unaffected by their operability.

Fire-induced failure of the added interlock between valve E12F004C and RHR "C" pump may prevent the RHR "C" pump from starting. This is not a concern, however, as the RHR "C" train is not required for safe shutdown of the plant incident to a fire (it provides a LPCI function only in mitigation of a LOCA). Failure of the added RHR "A" and/or "B" pump start permissive bypass switches for valves E12F066A and B due to a fire may prevent the RHR "A" and/or "B" pump from operating with suction from the spent fuel pool. This is also not a concern since the associated spent fuel pool cooling assist mode of operation of the RHR system is not required for safe shutdown of the plant incident to a fire.

The installation requirements for the ADHRS preserve the integrity of fire barriers and penetration seals so existing plant fire containment features are unaffected.

On the basis of the above, no significantly increased consequences of a fire are indicated by installation of the ADHRS.

Offsite Radioactive Release

The potential consequences of an offsite release of radioactively contaminated water due to an ADHRS heat exchanger tube failure have been shown to be within regulatory guidelines and limits and less than comparable events evaluated in the UFSAR, Section 15.7.2.

Inadvertent Operation of RHR Shutdown Cooling

As discussed in UFSAR Section 15.1.6, the effect of inadvertent operation of RHR is to slowly decrease reactor coolant temperature. Since the heat removal capacity of the RHR system is greater than that of ADHRS, the ADHRS cannot produce a greater effect. The mitigation of the event is a high neutron flux reactor scram, which would be unaffected by the installation or use of the ADHRS.

(c) The function of the PSW discharge line radiation monitor is to detect substantial intersystem leakage of reactor coolant into the PSW system. This monitor and its associated alarm performs no automatic accident mitigation function or safety-related function. The PSW radiation monitor is not required to function in order to prevent accidental offsite doses from being exceeded. The design and installation of the PSW radiation monitor will be done in accordance with appropriate codes and standards to ensure that interfacing system requirements are not compromised.

(d) Currently, valves E12F066A and B perform no accident mitigation function and are not required to open or close in order to bring the plant to cold shutdown conditions. The function of the two new motor operated valve thermal overload protection devices is to protect the motor operators under overload conditions. These motor operators perform no automatic or remote-manual accident mitigation function. The failure of the motor operators will have no functional effect on ECCS system operation. The ECCS system is capable of delivering required system flow rates with E12F066A and B failed in the open position. The failure of the motor operator to close or to be closed would prevent the establishment of an ASME Section III Class 2 pressure boundary for ECCS. However, the ECCS system has been evaluated with the failure of these valves in the open position and was found to be capable of continuing to deliver the required ECCS system flow rates. In addition, the piping beyond valves E12F066A and B is designed for process conditions that would occur in the LPCI mode. The only function of the motor operators is to allow remote-manual operation of the valves from the control room. The addition of the motor operators and the thermal overload devices will not change or affect the valves' present function or performance.

Therefore, the addition of the ADHRS and its associated PSW radiation monitor and thermal overload devices for valves E12F066A and B would not significantly increase the probability or consequences of an accident previously evaluated.

II. These changes would not create the possibility of a new or different kind of accident from any previously evaluated.

(a) Based on the functional interactions of the ADHRS, the following potential events have been evaluated with respect to the creation of a new or different type of accident from any previously evaluated:

o Loss of fuel pool cooling/fuel pool water inventory

o Offsite radioactive release

Loss of Fuel Pool Cooling/Fuel Pool Water Inventory

The ADHRS can be connected to the spent fuel pool either in a flush/fill mode or as a suction source for cooling when the reactor cavity/upper containment pool is filled. The ADHRS is not specifically designed for or otherwise expected to assume heat loads imposed by the spent fuel pool. Additionally, the loss or failure of the ADHRS cannot prevent the RHR system from operating in the fuel pool cooling mode. Thus, loss of the ADHRS cannot involve a loss of spent fuel pool cooling.

The operation of the ADHRS with suction on the spent fuel pool will not cause a loss of spent fuel pool cooling system, as described in UFSAR Section 9.1. There are no mechanical, controls, or electrical interconnections between the two systems and hydraulic-type effects have been evaluated and concluded to be non-interfering.

When connected to the spent fuel pool there is potential for leakage from the pool due to a failure in ADHRS piping. This piping is designed and qualified per the applicable criteria including new operating conditions for existing piping and is no more susceptible to a break or leak than existing piping. The potential for inadvertently draining the spent fuel pool when the ADHRS is taking suction from it has been evaluated. The ADHRS uses an existing suction point on the spent fuel pool. As explained in UFSAR Section 9.1, all connections to the pool are designed to preclude drainage below a level sufficient for adequate shielding. The ADHRS use of this existing line would not interfere with this feature. Hydraulic operation effects that could lower the spent fuel pool level have been evaluated and found not to present a potential hazard relative to the shielding function of the fuel pool inventory.

The drainage potential and hydraulic effects of ADHRS operation on the spent fuel pool are consistent with those existing when operating the RHR system in the fuel pool cooling assist mode. Since the ADHRS configuration offers no less protection in this regard, it is concluded that a new or different accident type is not created.

Offsite Radioactive Release

The ADHRS design provides a boundary, within the ADHRS heat exchangers, between radioactively contaminated reactor coolant and the PSW system which can discharge to the environment. The boundary (heat exchanger tubes) is designed and constructed to ASME Section III and Seismic Category I requirements. As such the probability of an ADHRS boundary failure is no greater than other plant systems containing radioactive fluids (e.g. FPCCU, RWCU, or radwaste tanks).

The GGNS UFSAR Section 15.7.2 addresses a postulated unexpected and uncontrolled release of radioactivity due to a radioactive liquid waste system failure. The postulated gross failure of the ADHRS pressure boundary and the UFSAR Section 15.7.2 event are considered similar kinds of accidents in that both are unexpected and uncontrolled releases of radioactive material to the offsite boundary.

(b) The PSW radiation monitor will perform no automatic accident mitigation function and will initiate no safety-related or nonsafety-related systems. The function of the radiation monitor is to detect substantial intersystem leakage of reactor coolant into the PSW system. The design and installation of the PSW radiation monitor will be done in accordance with appropriate codes and standards to ensure interfacing system requirements are not compromised.

(c) The addition of the motor operators and the thermal overload protection devices on valves E12F066A and B does not change the valves [SIC] original function. These valves will continue to perform no safety-related function other than serving as system code boundary classification break. The failure of E12F066A or B to close or to be closed by the operator defeats the establishment of an ASME Section III Class 2 pressure boundary for ECCS as recommended in Regulatory Guide 1.29 and committed to in UFSAR Section 3.2. However, this failure has no functional effect on the operation of ECCS. The ECCS systems have been evaluated with the failure of these valves in the open position and was [SIC] found to be capable of continuing to deliver the required ECCS flow rates. In addition, the piping beyond these valves is safety-related (Class 3), Seismic Category I and designed for the process conditions that would occur in the LPCI mode.

Therefore, the addition of the ADHRS and its associated PSW radiation monitor and thermal overload devices for valves E12F066A and B would not create the possibility of a new or different kind of accident from any previously evaluated.

III. These changes would not involve a significant reduction in the margin of safety.

(a) The ADHRS provides an improved alternate method of decay heat removal and coolant mixing as required by the bases for Technical Specification 3/4.4.9. Since the ADHRS is specifically designed to maintain the average reactor coolant temperature less than or equal to 200° F in cold shutdown and less than or equal to 140° F in refueling, the margin of safety to fulfill the requirements of Technical Specification 3/4.4.9 and Table 1.2 is maintained.

(b) The ADHRS is designed and constructed to ASME Section III, Class 2 and Class 3 standards and to Seismic Category I criteria. As such, the margin of safety this system provides is equal to or greater than other comparable plant systems containing radioactive fluids (e.g. FPCCU, RWCU, etc.). The ADHRS and its associated PSW radiation monitor will have no direct or indirect impact on existing safety-related or nonsafety-related systems and thus will not affect operation of any equipment required to mitigate an accident.

(c) The addition of the motor operators and associated thermal overload protection devices to valves E12F066A and B will not affect the operation or intended function of these valves. As previously noted, these valves perform no accident mitigation function. The current function of these valves is to align RHR "A" or "B" loops in the spent fuel pool cooling assist modes and to serve as code boundary classification breaks between ASME Section III Class 2 and Class 3 piping. The failure of these valve actuators will have no functional effect on the ECCS system.

Therefore, the addition of the ADHRS and its associated PSW radiation monitor and thermal overload devices for valves E12F066A and B would not involve a significant reduction in the margin of safety.

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

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

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

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

NRC Project Director: Elinor G. Adensam

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

Date of amendment request: December 2, 1988

Description of amendment request: The amendment would change the Technical Specifications (TS) by changing the surveillance requirements in TS 3/4.9.6.3, "Fuel Handling Platform," to accommodate the addition of a new auxiliary hoist. This new hoist will be used for handling control rods in the spent fuel pool.

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

The licensee has provided an analysis of no significant hazards considerations in its request for a license amendment. The licensee's analysis of the proposed amendment against the three standards in 10 CFR 50.92 is reproduced below.

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

a. The accident previously evaluated is a Fuel Handling Accident. The probability of a Fuel Handling Accident is not significantly increased because the components being lifted would be lifted by the monorail auxiliary hoist if there were not an auxiliary hoist. The auxiliary hoist is designed, manufactured and installed with the same appropriate criteria as the monorail auxiliary hoist. Additionally the auxiliary hoist for the FHP [fuel handling platform] is the same design, manufacture and installation as the auxiliary hoist on the [refueling platform] RP.

b. The consequences of a fuel handling accident are not increased due to the fact that no new loads or lifting heights are introduced with the addition of the auxiliary hoist on the FHP. Additionally the Fuel Handling Accident fuel damage bounds the scenario of a load dropping from the auxiliary hoist on the FHP.

c. The proposed changes in the Surveillance Requirements provide clarification to allow performance of the appropriate surveillance on the correct hoist. Thus there is not a significant increase in the probability or consequences of an accident previously evaluated involved in this change.

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

a. The fuel handling accident is the only accident possible. The fuel handling accident has been previously evaluated and bounds the fuel damage resulting from a load drop from the auxiliary hoist on the FHP.

b. The RP currently has an auxiliary hoist of the same manufacture, design and installation. The FHP auxiliary hoist will not move spent fuel or new fuel. The RP and FHP auxiliary hoist load is limited to less than the load used in the fuel handling accident and the load lift height is limited to the height of the lift allowed for the RP auxiliary hoist. The load drop from the auxiliary hoist of the FHP is bounded in fuel damage by the Fuel Handling Accident.

c. The proposed changes in the Surveillance Requirements will ensure that the correct hoist has the appropriate surveillance performed.

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

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

a. The design, manufacture, installation, load limits, up travel hoist limit, fuel handling exclusion and applicable interlocks define the margin of safety for the auxiliary hoist on the Fuel Handling platform. Since the FHP auxiliary hoist and the RP auxiliary hoist are not significantly different in these respects this change does not involve a significant reduction in the margin of safety.

b. The proposed change ensures that the appropriate surveillance is performed on the correct hoist. This will verify that the redundant load limits for both the auxiliary hoists on the FHP and the load override switch on the monorail auxiliary hoist are operable and are in the correct position.

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

Date of amendment request: October 13, 1988

Description of amendment request: The proposed amendment to the Technical Specifications (TS) will add a paragraph which is currently included in Bases 3.6.I to TS Section 4.6.I.1. This paragraph provides clarifying information for determining required inspection intervals and establishes criteria for snubbers that may be exempted from being counted as 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).

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 any accident previously evaluated.

The requirements for visual inspections of safety-related snubbers are not being changed.

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

The existing requirements for snubber surveillance are not being changed.

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

The level of safety provided in the existing Technical Specifications will be maintained.

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, Counselors at Law, City Place, Hartford, Connecticut 06103-3499.

NRC Project Director: John F. Stolz

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

Date of amendment request: December 2, 1988

Description of amendment request: The proposed amendment to the Technical Specifications will reflect the implementation of modifications related to degraded grid protection for Class 1E power systems. These modifications are scheduled to be completed during the 1989 refueling outage.

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).

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 any accident previously evaluated. The impact of the change on design basis accidents (DBAs) which assume loss of off-site power has been reviewed by the licensee and has been determined to be unaffected by the change. The proposed changes, in general, provide for more conservatism in that the operability requirements for the on-site emergency buses and their associated control circuits are more restrictive than the existing requirements. These changes to the Technical Specifications do not impact the failure probability of the associated electrical system, rather they increase the probability that a train of emergency electrical power is available following an accident. The probability of occurrence or the consequences of the DBAs are unchanged.

2. Create the possibility of a new or different kind of accident from any previously evaluated. There are no new failure modes associated with the proposed changes since they involve more restrictive requirements on the operability of the electrical power systems. No new accident is created because the same equipment is assumed to perform in the same manner as before.

3. Involve a significant reduction in a margin of safety. The protective boundaries are not impacted because the consequences of the DBA are not affected. Since the protective boundaries are not affected, the safety limits are also not affected. The proposed change maintains the basis of the Technical Specifications in assuring electrical power operability.

The existing 345 KV loss of normal power (LNP) sensing inputs are not in the Technical Specifications. Technical Specification for the new 4160 volt bus undervoltage/timing relay input circuits will be added. The existing Technical Specification for the bus voltage permissive and the LNP actuation circuits are being modified to provide additional detail and clarification.

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, Counselors at Law, City Place, Hartford, Connecticut 06103-3499.

NRC Project Director: John F. Stolz

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

Date of amendment request: December 31, 1988.

Description of amendment request: This proposed amendment would revise the Technical Specifications (TS) to change the minimum operating requirements for the Raw Water Pumps to allow operation with one inoperable raw water pump when river water temperature is below 60° F.

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. The licensee addressed the above three standards in the amendment application.

With regard to the three standards, the licensee states that operation of the facility in accordance with this amendment would not:

(a) Involve a significant increase in the probability or consequences of an accident previously evaluated. This change decreases the consequences of a loss of coolant accident or main steam line break event with concurrent loss of off-site power and failure of a diesel generator by ensuring that the closed cooling water heat exchangers have sufficient heat removal capacity to maintain the containment below the design basis maximum pressure.

(b) Create the possibility of a new or different kind of accident from any accident previously evaluated. This change does not propose new or different modes of operation for the plant. The continued use of the same Technical Specification administrative controls prevents the possibility of a new or different kind of accident.

(c) Involve a significant reduction in a margin of safety. This change increases the minimum operability requirements of the raw water and containment cooling system and, therefore, will not cause any reduction in the margin of safety.

The staff has reviewed the licensee's no significant hazards consideration determination and agrees with the analysis. In addition, the Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing examples (51 FR 7751) of amendments that are considered not likely to involve significant hazards consideration. The proposed change to minimum operating requirements for the raw water pumps is an additional restriction to the allowable minimum requirements presently in the Technical Specifications and is, thus, similar to the example of changes which constitute an additional limitation, restriction, or control not presently included in the technical specifications.

Accordingly, the staff proposes to determine that the proposed changes to the Technical Specification do not involve a significant hazards consideration.

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

Attorney for licensee: LeBoeuf, Lamb, Leiby, and MacRae, 1333 New Hampshire Avenue, NW., Washington, 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 9, 1988

Description of amendment request: The proposed amendment revises the Emergency Diesel Generator (EDG) Technical Specifications (TS) 3.8.1.1, 3.8.1.2 and BASES. The proposed changes represent the recommendations of the diesel generator manufacturer and are consistent with the appropriate recommendations provided in the Commission's Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability", dated July 2, 1984. The proposed changes will improve the reliability of the EDGs and reduce the risk from possible station blackout.

Generic Letter 84-15 discussed the need to assure that the reliability of the EDGs is maintained at an acceptable level. This would be accomplished in part by reducing the number of "cold fast starts" or demonstrating starting capabilities from ambient conditions with full electrical loads without the considerations of prelubrication or prewarming prior to the test. The staff concluded that cold fast starts cause unnecessary premature diesel engine degradation and that some testing techniques for EDGs did not take into consideration their manufacturers recommended preparatory actions such as prelubrication of moving parts and warm-up procedures prior to starting.

Specifically the licensee proposes to change the TS to:

(A) Specify prelubrication and prewarming of EDGs prior to preplanned EDG starts. Prelubrication/prewarming would decrease the wear and stress on the EDGs and would increase reliability and availability.

(B) Provide for gradual acceleration and gradual electrical load increases to an indicated load range during EDG testing in accordance with manufacturers recommendations in order to decrease the stresses inherent with rapid acceleration, sudden large electrical load changes and routine overloading.

(C) Revise the surveillance starting/testing frequency in order to limit the incremental wear and stress on the EDGs.

(D) Revise the accelerated starting test frequency program. The program will be based upon the number of failures in the last 20 demands in lieu of the failures in the last 100 valid tests in order to maintain increased EDG reliability without excessive and damaging surveillance cold fast starts.

(E) Allow EDG maintenance inspections (18 month tear down) while at power, rather than "during shutdown", in order to decrease outage time used for maintenance tear down and allow more time for inspection in lieu of the requirement for tear down inspections only under a limited outage schedule.

(F) Incorporate the 184 day starting surveillance into Surveillance 4.8.1.1.2.h. allowing this new paragraph for prelubrication and prewarming of the EDGs with electrical loading to an indicated range within 200 seconds.

(G) Eliminate from Specification 4.8.1.1.2.a the need for a staggered test basis.

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, (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 as to whether the proposed amendment involves a significant hazards consideration. The licensee's analysis is summarized as follows:

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

The proposed changes only revise surveillance test methods and schedules. The proposed changes incorporate the recommendations of the diesel generator manufacturer and the appropriate recommendations provided in Generic Letter 84-15 to improve the reliability and availability of the EDGs. These changes do not modify or add any initiating parameters that would significantly increase the probability or consequences of any previously analyzed accident. In fact, the proposed changes would reduce the unnecessary wear and stress on the EDGs and would provide the capability to perform various surveillance requirements during operation.

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

As discussed in Item (1) above, the proposed changes only revise surveillance test methods and schedules that will improve the reliability and availability of the EDGs. The operation and/or design of the onsite emergency power system is not being changed and no physical plant modification is involved. As such, the plant initial conditions utilized for the design basis accident analysis remain valid and no new or different kind of accident is created.

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

As discussed in Item (1) above, decreasing the harsh and potentially damaging manner of testing of the EDGs would lead to greater reliability and availability. The proposed surveillance testing and frequency would continue to ensure the availability of the EDGs consistent with the previous Commission guidance and thus will not involve a reduction in the margin of safety.

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

Local Public Document Room location: 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

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

Date of amendment request: December 14, 1988

Description of amendment request: The proposed amendment would change the Technical Specifications (TSs) to permit removal of the Rod Sequence Control System (RSCS) and to reduce the Rod Worth Minimizer (RWM) low power setpoint.

Basis for proposed no significant hazards consideration determination: The Rod Sequence Control System restricts rod movement to minimize the individual worth of control rods to lessen the consequences of a Rod Drop Accident (RDA). Control rod movement is restricted through the use of rod select, insert, and withdrawal blocks. The Rod Sequence Control System is a hardwired (as opposed to a computer controlled), redundant backup to the Rod Worth Minimizer. It is independent of the Rod Worth Minimizer in terms of inputs and outputs but the two systems are compatible. The RSCS is designed to monitor and block when necessary operator control rod selection, withdrawal and insertion actions, and thus assist in preventing significant control rod pattern errors which could lead to a control rod with a high reactivity worth (if dropped). A significant pattern error is one of several abnormal events all of which must occur to have a RDA which might exceed fuel energy density limit criteria for the event. It was designed only for possible mitigation of the RDA and is active only during low power operation (currently generally less than 20 percent power) when a RDA might be significant. It provides rod blocks on detection of a significant pattern error. It does not prevent a RDA. A similar pattern control function is also performed by the RWM, a computer controlled system. All reactors having a RSCS also have a RWM.

In response to a topical report submitted by the BWR Owner's Group, on December 27, 1987 we issued a letter with a supporting safety evaluation approving (1) elimination of the RSCS while retaining the RWM to provide backup to the operator for control rod pattern control and (2) lowering the setpoint for turnoff of RWM to 10% of rated thermal power from its current 20% level. (Letter, A. C. Thadani, NRC to J. S. Charnley, CE, Subject: Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Revision 8, Amendment 17.)

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 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 December 14, 1988 letter states the following in response to the three NRC criteria referenced above:

Deletion of the RSCS and reduction of the RWMS low power set point from 20% to 10% does not involve a significant hazards consideration because it does not:

1. Significantly increase the probability of occurrence or the consequences of an accident or malfunction of equipment related to safety as previously evaluated in the FSAR.

The nonsafety-related RSCS and RWM are not required for nor do they support the proper operation of other systems. Hence, deleting the RSCS and changing the low power set point on the RWM has no effect one way or the other on the probability of equipment malfunction in other systems or within the RWM.

The probability of occurrence of an accident is not affected by this change. These changes impact only the rod drop accident (RDA) analyses. The probability of an RDA is dependent only on the control rod drive system and mechanisms themselves, and not in any way on the RSCS or RWM.

The consequence of an RDA as evaluated in the LGS FSAR will not be affected by this modification. An extensive probabilistic study was performed by the NRC staff (letter and enclosure from B. C. Rusche, NRR, to R. Fraley, ACRS, dated June 1, 1976, "Generic Item II A-2 Control Rod Drop Accident (BWRs)"). This study indicated that there was not a need for the RSCS. Furthermore, improved methodologies in the RDA analysis methods (e.g. BNL-NUREG 28109, "Thermal Hydraulic Effects on Center Rod Drop Accident in a BWR," October 1980) indicated that the peak fuel enthalpies resulting from an RDA are significantly lower than previously determined by less refined methodologies.

The RSCS duplicates the function of the RWM. So long as the RWM is operable, the RSCS is not needed since the RWM prevents control rod pattern error. In the event the RWM is out of service, after the withdrawal of the first 12 control rods, the proposed Technical Specifications require that control rod movement and compliance with the prescribed control rod pattern be verified by a second licensed operator or technically qualified member of the technical staff. The verification process is controlled procedurally to ensure a high quality, independent review of control rod movement. In addition, to further minimize control rod movement at low power with the RWM out of service, the proposed Technical Specifications will permit only one plant start-up per calendar year with the RWM out of service prior to or during the withdrawal of the first twelve control rods. All the above taken together demonstrates consistency and applicability to those conclusions reached in the referenced SER, and substantiate the conclusion that there will be no increase in the consequences of an RDA as evaluated in the FSAR as a result of eliminating the RSCS.

There will also be no increase in the consequences of an RDA as evaluated in the FSAR due to lowering the RWM set point from 20% to 10%. The effects of an RDA are more severe at low power levels and are less severe as power level increases. Although the original calculations for the RDA were performed at 10% power, the NRC required that the generic BWR Technical Specifications be written to require operation of the RWM below 20% power in order to ensure conservatism. However, GE continued to perform the RDA analyses at and below 10% power because these produced more conservative analytical results. Recently, more refined calculations by BNL have shown that even with the maximum single control rod position error, and most multiple control rod error patterns, the peak fuel rod enthalpy reached during an RDA from these control rod patterns would not exceed the NRC limit of 280 cal/gm for RDAs above 10% power, confirming the original GE analyses. Hence, lowering the RWM set point from 20% to 10% will not result in an increase in the consequences of an RDA as evaluated in the FSAR. The previously referenced NRC SER has concluded this RWM set point reduction to be acceptable.

2. Create the possibility for an accident or malfunction of a different type than any evaluated in the FSAR.

Operation of the RSCS and RWM cannot cause or prevent an accident. They function to minimize the consequences of an RDA. The RDA is already evaluated in the FSAR, and the effect of this proposed change on the analyses is discussed in Item 1 above.

Elimination of the RSCS and lowering the RWM set point will have no impact on the operation of any other systems, and hence would not contribute to a malfunction in any other equipment nor create the possibility for an accident to occur which has not already been evaluated.

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

Elimination of the RSCS will not lower the margin of safety for the reasons discussed in Item 1 above and summarized below:

(a) An extensive NRC study has determined that the possibility of an RDA resulting in unacceptable consequences is so low as to negate the requirement for the RSCS.

(b) Recent calculations have determined that the consequences of an RDA are acceptable above 10% power.

(c) The RSCS is redundant in function to the RWM. Eliminating the RSCS does not eliminate the control rod pattern monitoring function performed by the RWM.

(d) To ensure that the RWM will be in service when required, the proposed RWM Technical Specification will be revised to allow only one startup per calendar year with the RWM out of service prior to or during the withdrawal of the first twelve control rods. If the RWM is out of service below 10% power, control rod movement and compliance with prescribed control rod patterns will be verified by a second licensed operator or technically qualified member of the technical staff. This situation is controlled by a station procedure which specifically requires the following:

* Plant Management approval is required in order for the operator to bypass the inoperable RWM.
* A second operator or technically qualified staff member, with no other duties, is required to verify the first operator's actions while the first operator performs rod movements.
* The startup and the shutdown sequences with their respective signoff sheets are provided for verification by the second operator for each step and rod movement made by the first operator.
* The operators are provided with shutdown instructions that would allow a shutdown sequence that would result in a control rod pattern which would agree with RWM pattern if that system were not bypassed and was controlling. These instructions identify, for the operator, the RWM shutdown step to be initiated for further rod insertion below the RWM low power setpoint (i.e. 10% reactor power).

There is no significant reduction in the margin of safety resulting from lowering the RWM set point from 20% to 10% because calculations by GE and BNL have shown that even with the maximum single control rod position error, and most multiple error patterns, the peak fuel rod enthalpy during an RDA from these patterns would not exceed the NRC limit (280cal/gm) above 10% power.

In summary GE has provided technical justification for the proposed changes in the Topical Report NEDE-24011-P-A and associated references which justify the acceptability of the proposed changes.

The NRC has reviewed and accepted the GE analysis and provided guidelines for licensees wanting to make the changes proposed in NEDE-24011-P-A and approved in the NRC SER issued December 27, 1987 to J. S. Charnley of General Electric.

The proposed changes are consistent with those approved in the NRC SER and the guidelines setforth therein. Therefore, there is no significant reduction in a margin of safety.

The staff has reviewed the licensee's submittal and significant hazards analysis and concurs with the licensee's determination as to whether the proposed amendment involves a significant hazards consideration. Therefore, the staff proposes to determine that the proposed amendment involves no 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. 50-286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: December 8, 1988

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

Technical Specification 3.4.B provides that if the requirement of three operable auxiliary feedwater pumps cannot be met within 72 hours, the reactor shall be in hot shutdown within the next 12 hours. This wording renders the specification applicable to all possible conditions, independent of the number of the inoperable auxiliary feedwater pumps. The Westinghouse Standard Technical Specifications provide LCOs which are dependent on the number of inoperable auxiliary feedwater pumps. The proposed change will revise Technical Specification 3.4 to reflect the applicable LCOs provided by the Westinghouse Standard 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; 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 made the following analysis:

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

Response

The proposed amendment involves a revision to the current LCO for the auxiliary feedwater pumps to reflect the applicable LCOs provided by the Westinghouse Standard Technical Specifications. The proposed amendment will provide a LCO based on the number of inoperable pumps. The proposed LCOs for the situations of one and two pumps inoperable are equivalent or more stringent than the existing LCO. The proposed LCO for the situation of three pumps inoperable reflects the fact that continued plant operations with three inoperable auxiliary feedwater pumps is a safer mode of operation than commencing plant shutdown in such a condition. As such, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response

The proposed amendment does not involve any physical alteration to the auxiliary feedwater system or to any other plant system or structure. The change does not affect the operation of any plant system. Hence, the possibility of a new or different kind of accident from any accident previously evaluated is not created by this change.

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

Response

The proposed amendment involves a revision to the current LCO for the auxiliary feedwater pumps to reflect the applicable LCOs provided by the Westinghouse Standard Technical Specifications. The proposed amendment will provide a LCO based on the number of inoperable pumps. The proposed LCOs for the situations of one and two inoperable pumps are equivalent or more stringent than the existing LCO. The proposed LCO for the situation of three inoperable pumps reflects the fact that continued plant operations with three inoperable auxiliary feedwater pumps is a safer mode of operation than commencing plant shutdown in such a condition. As such, the proposed amendment does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Robert A. Capra, Director

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

Date of amendment request: December 19, 1988

Description of amendment request: As reported in LER 88-012-01, dated September 21, 1988 (P-88345), an error had been discovered by General Atomics in their computer program code utilized in the development of the environmental qualification profiles for the Fort St. Vrain (FSV) equipment qualification (EQ) program. The error resulted in generating nonconservative building temperatures for a variety of smaller high energy line break (HELB) sizes. Reanalyses of the HELB scenarios confirmed that a number of smaller HELB sizes resulted in higher delayed building temperature peaks than previously analyzed. To reduce those peaks, a modification to the Steamline Rupture Detection/Isolation Setpoint was required. The new Fixed High Temperature Trip Setpoint is based on an Analysis Value of 180 degrees fahrenheit. This trip will allow automatic isolation of those HELBs which do not produce a building temperature rate-of-rise rapid enough to cause a trip of SLRDIS. The Fixed High Temperature Trip Setpoint enhances equipment qualification by reducing the magnitude of the delayed building temperature peaks for smaller HELBs and permits timely operator access to perform any manual actions which may be required. The final EQ composite temperature profile used for future equipment qualification incorporates the new Fixed High Temperature Trip Setpoint in addition to the existing Rate-of-Rise Trip Setpoint as submitted in P-88344 dated September 28, 1988.

This change to the Technical Specifications implements this Fixed High Temperature Trip Setpoint by modifying LCO 4.41 and SR 5.4.1.

Basis for proposed no significant hazards consideration determination: The licensee submitted the following evaluation of significant hazards considerations for this amendment request:

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

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

Recent reanalyses determined that certain postulated HELBs could result in temperatures in excess of the original composite temperature profile used in the Environmental Qualification Program. While the initial peak temperatures were reduced, a variety of smaller breaks resulted in higher delayed temperatures. The addition of a new Fixed High Temperature Trip Setpoint will enable SLRDIS to automatically detect and isolate a number of the smaller HELBs which do not produce a temperature rate-of-rise rapid enough to cause a SLRDIS Rate-of-Rise Trip. This trip would significantly mitigate the delayed temperature peaks, thus enhancing equipment qualification and ensuring that the resultant temperatures will not prevent timely building access. The electrical equipment per 10 CFR 50.49 remains qualified with the new composite profile. The consequences of a SLRDIS actuation are addressed in FSAR Section 7.3.10.4. The consequences of SLRDIS actuations are no different than previously analyzed for the Rate-of-Rise Trip Setpoint.

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

The required function of SLRDIS is to isolate both primary and secondary coolant loops in the event of a HELB, resulting in an interruption of forced circulation cooling (IOFC). The consequences of a SLRDIS actuation have been evaluated in the FSAR. The added Fixed High Temperature Trip Setpoint adds a new point in the elevation of bulk building average temperature at which SLRDIS will trip.

Since the new trip results only in changing the programming, all previous failure modes and malfunctions remain unchanged. Recovery following a SLRDIS actuation may be accomplished by one of three methods: (1) normal keyboard commands provided that the ambient temperature sensed by at least 3 of the 4 sensors has returned below the trip setpoint; or (2) manipulating the SLRDIS detection rack bypass key switch; or (3) removal of the SLRDIS XCR's (control relays) in control board 93-I-10.

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

The basis for SLRDIS in LCO 4.4.1 states that detection of steam leaks is required for environmental qualification of safe shutdown cooling systems. The detection system utilizes a pre-trip fixed temperature alarm to initiate manual operator actions to isolate the steam leak and a Rate-of-Rise Trip Setpoints to initiate an automatic isolation. Both detection points ensure that the resultant harsh environment from a postulated steam leak is limited such that the safe shutdown systems will be capable of performing their safety function under the environmental conditions and permit timely building access. Adding a new Fixed High Temperature Trip Setpoint enhances the required detection and resultant protective actions to limit the harsh environment. The same system accuracy and setpoint uncertainty methodology were applied to calculate the new Fixed High Temperature Allowable Value and Trip Setpoint. The added trip setpoint neither causes any new single failure points nor compromises the system detection. The operability requirements as defined in the Technical Specifications remain unchanged for the new HELB analyses and SLRDIS Fixed High Temperature Trip Setpoint.

No single failure can result in the actuation of the SLRDIS safety function or preclude the safety function from occurring. EE-EQ-0033, Rev. C has been updated to conservatively reflect the impact of a prolonged loss of HVAC condition on SLRDIS actuation with the new Fixed High Temperature Trip Setpoint. The temperature in either the reactor building or turbine building will not cause a SLRDIS actuation for a loss of HVAC condition of less than or equal to 6 hours. Prior to 6 hours, corrective action, if required, could be implemented to reduce building temperatures. The new trip setpoint is slightly higher than the existing reactor building high temperature scram setpoint of 161 degrees F, which has not actuated in the past and would not be expected to actuate except during a HELB or Design Basis Accident No. 2. Therefore, it is concluded that the probability for inadvertent actuation of SLRDIS is not increased by the 171 degrees F. Fixed High Temperature Trip Setpoint.

The staff has reviewed the licensee's evaluation and agrees with its conclusions. Therefore, 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

Rochester Gas and Electric Corporation, Docket No. 50-244, R.E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: October 10, 1988 as supplemented on December 22, 1988

Description of Amendment: The proposed amendment would modify the Technical Specifications to reflect the appropriate titles for corporate positions having responsibility for overall plant safety, and to modify the PORC membership.

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

Proposed revisions to the Technical Specifications include an additional member of the Plant Operations Review Committee, Maintenance Planning/Scheduling Manager, and changes to the following titles: Senior Vice President to Vice President, Training Coordinator to Division Training Manager, Shift Foreman to Shift Supervisor, Plant Health Physicist to plant Health Physicist, and Electric and Steam Production to Production and Engineering. The license also requests addition of the position of Plant Manager, Ginna Station. This position has the responsibility for overall onsite operations of Ginna Station and replaces the titles of Superintendent, Ginna Production; and Station Superintendent, except that the position of Superintendent, Ginna Production continues as chairman of the Plant Operations Review Committee (PORC).

The changes are administrative in nature because they do not change the physical aspects of the plant, equipment, or previously approved operations, and they do not involve a significant increase in the probability or consequences of an accident previously evaluated; or create the possibility of a new or different kind of accident from any accident previously evaluated; or involve a significant reduction in the margin of safety. Accordingly, the Commission proposes to determine that this revision does not involve a significant hazard.

Local Public Document Room location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610.

Attorney for licensee: Harry Voigt, Le Boeuf, Lamb, Leiby and McRae, Suite 110, 1133 New Hampshire Avenue, NW., Washington, DC 20036.

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

Date of amendment requests: December 9, 1988 (TS 262)

Description of amendment requests: The proposed amendment would change the BFN Technical Specifications (TS) for Units 1, 2, and 3 to add requirement 1.0.MM.8 to the Definitions Section. This requirement would ensure that an Inservice Inspection Program for piping identified in NRC Generic Letter 88-01 be performed in accordance with developed NRC positions.

Specifically, this proposed change would require the implementation of a program to monitor piping made of austenitic stainless steel meeting the operational requirements outlined in Generic Letter 88-01 for any indications of IGSCC. This proposed change has been reviewed and approved by the BFN Plant Operations Review Committee and the Nuclear Safety Review Board.

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

NRC 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 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 of consequences of an accident previously evaluated, 2) create the possibility of a new or different kind of accident from an accident previously evaluated, or 3) involve a significant reduction in a margin of safety.

1. The proposed change does not involve a significant increase in the probability or consequence of any accident previously evaluated.

BFN Final Safety Analysis Report (FSAR) Chapter 14 provides analysis of Design Basis Accidents (DBA) in which BFN was analyzed and licensed. In reviewing these DBA's the one closest to the IGSCC issue would be the Loss of Coolant Accident (LOCA) discussed in FSAR Section 14.6.3. In that analysis, it is assumed that the reactor is operating at the most severe condition at the time the recirculation pipe breaks, which would maximize the parameter of interest: primary containment response, fission product release or Core Standby Cooling System requirements. In addition, the recirculation loop pipeline is assumed to be instantly severed. This results in the most rapid coolant loss and depressurization with coolant discharged from both ends of the break.

The IGSCC concern of the staff results from various BWR Plants identifying that they have experienced some cracking in weldments of austenitic stainless steel piping. The proposed TS would require an inspection program be performed in accordance with NRC guidelines to ensure that the potential of pipe weldment cracking be minimized. This TS would apply to all BWR piping made of austenitic stainless steel that is four inches or larger in nominal diameter and contains reactor coolant at a temperature above 200° F during power operation regardless of Code Classification. Implementation of this TS would ensure that weldment cracking would be detected and fixed before a pipe would rupture. As a result, the proposed TS would provide added assurance of not exceeding any assumptions or results for the LOCA analysis stated above.

In Generic Letter 88-01, the Commission stated that unless appropriate remedial actions are taken, BWR plants may not be in compliance with their current design and licensing bases, including 10 CFR 50, Appendix A, General Design Criteria (GDC) 4, 14, and 31. NRC proposes a TS, in which BFN is hereby proposing, which will ensure implementation of a NRC approved program for IGSCC. This program provides appropriate remedial actions, therefore, providing added assurance that BFN is within its design bases and appropriate 10 CFR 50 Appendix A GDC's.

2. The proposed change does not change or modify the operation or design of any safety-related equipment currently installed at BFN. This proposed TS would enhance the overall plant integrity through the implementation of a program that would provide added assurance that the pressure boundary piping integrity would be maintained. This proposed TS change is administrative in nature and does not introduce any new conditions which would create a new or different accident that has been previously analyzed.

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

The proposed change is administrative in nature and in fact enhances the margin of safety at BFN. Implementation of a program in accordance with Generic Letter 88-01 requires an ISI program to monitor specific piping that may be susceptible to IGSCC. This program would assist in detection of weldment cracking in austenitic stainless steel piping as outlined in the subject letter. The addition of this program provides added assurance that any cracking would be detected and fixed therefore, eliminating any added potential of a pipe rupture. The implementation of this TS enhances the overall integrity and safety of BFN. This proposed change also supports the design and licensing bases of BFN, in addition to supporting 10 CFR 50, Appendix A, GDC 4, 14, and 31.

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

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

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

NRC Assistant Director: Suzanne Black

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

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

Description of amendment requests: The Tennessee Valley Authority (TVA or the licensee) proposes to amend the Browns Ferry Technical Specifications (TS) by adding surveillance requirements to the Reactor Protection System (RPS) power monitoring system. In response to a NRC directive, the licensee installed RPS power monitoring system circuit protectors. The proposed surveillance requirements relate to these modifications.

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

1. The proposed change does not involve a significant increase in the probability or consequence of any accident previously evaluated.

A modification was made to install redundant class 1E circuit protection devices between the non-class 1E RPS power supplies and the class 1E RPS power supplies. These circuit protective devices consist of a contractor which will open by (1) an overvoltage relay with a trip level setting of less than or equal to 126.5 Vac, (2) an undervoltage relay with a trip level setting of greater than or equal to 113.4 Vac for the MG sets, (3) an undervoltage relay with a trip level setting of greater than or equal to 111.8 Vac for alternate supply, and (4) an underfrequency relay trip level setting of greater than or equal to 57 Hz on all devices.

The cabinets and conduits for each RPS power monitoring system are located in the control building, which is a seismic category 1 structure.

This structure will provide protection from effects of tornadoes, tornado missiles, and external floods. The components of each monitoring system are also seismically qualified for class 1E application as required by GDC 2.

In order to comply to GDC 21, there are two physically independent and fully redundant circuit interrupters provided for each RPS bus, including alternate supply. This redundancy provides single failure protection in case one circuit does not function properly. This also provides sufficient reliability to ensure the RPS performs its intended safety function.

The BFN Final Safety Analysis (FSAR) Section 7.2.3.2 states that the power to each of the two reactor protection trip systems is supplied, via a separate bus, by its own high-inertia, a-c motor generator set. The high inertia is provided by a flywheel. The inertia is sufficient to maintain voltage and frequency within 275% of rates values for at least 1.0 second following total loss of power to the MG set. In applying this to Section 14.5.4.4.b of the FSAR accident analysis, loss of auxiliary power assumes the RPS MG set coastdown time until loss of MG generator output voltage to be 5.0 seconds. Thus the upper and lower bounds for voltage output and time delay are identified as significant performance parameters expected from the MG set design. The RPS power monitoring system installed is designed by the MG sets to provide no time delay. Consequently, the trip level settings for the RPS power monitor must be outside the expected operating range of the MG set. For a nominal 120 Vac MG output voltage, the 5% regulation band (114 to 126 volts) is within the technical specification trip level setting of 113.4 to 126.5 Vac. This will allow the MG set to function within its intended and designed time and voltage range before the RPS power monitoring system trips. These settings support the design and function of the high-inertia MG sets, and therefore, support the assumptions made in the BFN FSAR.

Therefore, the design, trip level settings, and intended function of the RPS power monitoring system are both bounded and support the current BFN FSAR accident analysis.

2. The proposed amendment will not create the possibility of a new or different kind of accident.

The proposed change does not affect the operation or intended function of any currently installed safety related equipment. If all the protective circuits in one MG fail to open, the redundant train of RPS systems is still available to mitigate any design basis accident. The RPS power monitoring system does not perform any specific safety function therefore, failure would, at worst case, be bounded by the current BFN Final Safety Analysis.

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

The additional surveillance requirements resulting from the subject modification, enhance to overall dependability of the RPS system. By specifying overvoltage, undervoltage, and underfrequency values ensures that the RPS power monitoring system will protect the RPS components so they perform their intended function.

This system provides no direct safety function. It provides isolation between the non-class 1E RPS power supplies and the class 1E power distribution buses. It functions to isolate the RPS power distribution buses upon detection of overvoltage, undervoltage, and underfrequency on the RPS power supplies thereby preventing possible adverse operation of the class 1E RPS components outside their designed voltage and current ranges.

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

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

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

NRC Assistant Director: Suzanne Black

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 amendment request: November 28, 1988 as amended December 29, 1988.

Description of amendment request: The proposed amendment would increase the minimum critical power ratio (MCPR) in the Technical Specifications (TS) from 1.06 to 1.07 to account for additional uncertainties which normally occur in the second and subsequent operating cycles. It also would add two limiting lattice MAPLHGR (most-limiting average planar linear heat generation rate) curves to the TS to account for two new GE8X8EB fuel types being used this cycle and delete the MAPLHGR curve for the natural uranium bundles which are being totally removed during the refueling outage. The limiting conditions for operation and action statements for the APLHGR would be revised to reflect the lattice-dependent MAPLHGR limits in the GESTAR analysis and the default limits in TS figures for hand calculations. The TS would be reworded to clarify how power-dependent MAPLHGR factors ($MAPFAC_p$) are applied to the lattice MAPLHGR's and reflect NRC guidance in the GESTAR safety evaluation report. Figures and pages would also be renumbered and reordered. The extrapolated value for $MCPR_f$ (figure 3.2.2-1 of the TS) (the calculated MCPR at a given point of core flow) is being corrected by the factor of [1.0 +0.0032 (40-F)] where F = percent of rated core flow, to provide additional conservatism at lower core flows (below 40% of rated flow). Additionally, the flow dependent MAPLHGR factor ($MAPFAC_f$) curve (figure 3.2.1-4) is being extended down to the 20% rated core flow line to account for core flow shortfalls which were demonstrated during the startup testing program. (This figure is also being renumbered, the new number will be 3.2.1-1).

The proposed amendment would also delete two curves (A-A' and B-B') from the current set of MCPR parametric ($MCPR_p$) curves (those curves which specify the power-dependent MCPR limit at reduced feedwater temperatures for various core average exposures and core flows). Thermal margin during the second cycle will be maintained using curve C-C' as the limit and the C-C' designation will be removed. Additionally, the TS related to the LHGR (linear heat generation rate or heat-generation-per-unit-length of fuel rod) would be revised to reflect the higher LHGR associated with the new fuel. The definition of critical power ratio would be generalized by changing CEXL to a GE critical power correlation. The associated bases would also be changed.

Basis for proposed no significant hazards consideration determination: The standards used to arrive at a determination that a request for amendment requires no significant hazards consideration are included in the Commission's Regulations, 10 CFR 50.92(c), which states that the operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (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 licensees have provided the following analysis concerning the above three factors:

1. The proposed changes would not involve a significant increase in the consequences of an accident previously evaluated because MCPR, $MCPR_f$, LHGR, $MAPFAC_f$ and MAPLHGR core operating limits are provided to establish bounds on normal reactor operations which ensure core conditions are maintained within the assumptions and scope of accident analyses. New MAPLHGR and $MAPFAC_f$ curves and LHGR, MCPR and $MCPR_f$ limits are provided to reflect changes in the reactor fuel configuration and design assumptions. Operation within these limits will assure the consequences of affected transients and accidents will remain within the results and bounds of the safety analyses. MAPLHGR, $MAPFAC_f$, $MCPR_f$ and MCPR curves/limits were generated using analytical methods previously approved by the NRC.

The Safety Limit MCPR is determined using a statistical model that combines uncertainties in operating parameters with uncertainties used to calculate the critical power. For reload cores, some of the uncertainties used in the determination of the Safety Limit MCPR are larger than for initial cores. The higher Safety Limit MCPR for reload cores accounts for these increased uncertainties. The Safety Limit MCPR for reload cores has received previous NRC approval and is documented within the Updated Safety Analysis Report (USAR) and in GESTAR.

The critical power ratios in the transient analyses were calculated using the improved critical power correlation, GEXL-Plus. This correlation has been approved by the NRC (SER to Amendment 15 of GESTAR). Its predecessor GEXL was used for the initial core analysis.

The LHGR limit for GE8X8EB fuel was calculated using the GESTR-MECHANICAL code (a fuel rod thermal-mechanical performance model accepted by the NRC in GESTAR). Results from GESTR-MECHANICAL demonstrate that compliance with the new LHGR limit (in concert with appropriate MAPLHGR curves) will further ensure fuel design basis criteria are satisfied for GE8X8EB fuel.

Extended operating domains and modes of operation have been analyzed by GE (using NRC-approved methods) to determine applicable operating restrictions. GE demonstrated that the consequences of changes to the allowable operating region are bounded by the proposed values for MCPR and MAPLHGR. Furthermore, the probability of an accident is not increased because operation in the expanded region does not significantly alter the normal operation of the equipment, for which failures have been previously analyzed.

2. The proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated because proposed MCPR, $MCPR_f$, MAPLHGR, $MAPFAC_f$ and LHGR limits do not directly affect the operation or function of any system or component but instead set fuel or core thermal limits so that operation within these limits will maintain the analyzed margins of safety.

The Safety Limit MCPR was adjusted in the conservative direction because of calculational uncertainties associated with reload cores to maintain the margin of safety. MAPLHGR limits are provided for each bundle type to ensure that the requirements of 10 CFR 50.46 and Appendix K are maintained. The limits are results of the reload transient and Emergency Core Cooling System (ECCS) analyses and are designed to maintain the same margins of safety. Therefore, this change would not create the possibility of an accident different than previously evaluated.

Expanded operating regions represent changes to the core power and flow distribution, but do not significantly affect the operation or function of any system or component. Operating limits were established by analyses to bound all combinations of specified expanded operating domains and equipment out of service within acceptable analyzed conditions to ensure fuel integrity and ECCS criteria. Consequently, there is no significant impact on or addition to any system or equipment whose failure could initiate an accident.

3. The proposed changes would not involve a significant reduction in the margin of safety because all of the proposed changes have been analyzed to the same governing criteria as before and demonstrate that the consequences of transients or accidents are not increased beyond these already evaluated by the NRC for the Perry Nuclear Power Plant.

The Safety Limit MCPR is set at the point at which no fuel damage is expected to occur as discussed in GESTAR. The Safety Limit MCPR is combined with the most limiting transient change to the critical power ratio to establish the operating limit MCPR. The Safety Limit MCPR and the change resulting from the most limiting transient have been calculated by methods described in GESTAR. These methods have received previous NRC approval.

MAPLHGR's are determined by analysis to ensure the acceptance criteria of 10 CFR 50.46 are met and establish the margins of safety for fuel and the ECCS. Calculations using NRC-approved models described in GESTAR yield results within these acceptance criteria.

Furthermore, the fuel used in Cycle 2 is very similar to that used in the previous cycle and the core will be operated using NRC-approved methods.

The staff has reviewed the licensees' no significant hazards consideration determination. With respect to operation in the extended operating domains, the staff notes that the licensees have implemented procedures, in response to NRC Bulletin 88-07 dated June 15, 1988, which require scramming the plant immediately following a dual recirculation pump trip. Further, while in the maximum extended operating domain region, core flow will be maintained above 45%. These measures significantly reduce the possibility of power oscillations while in the extended operating domain. Given these considerations, the staff concurs with the licensees' no significant hazards determination. Therefore, the staff proposes to determine that the proposed amendment involves no significant hazards considerations.

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

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts & 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.

Commonwealth Edison Company, Docket No. 50-237, Dresden Nuclear Power Station, Unit No. 2, Grundy County, Illinois

Date of application for amendments: August 15, 1988

Brief description of amendments: The amendment revises the Technical Specifications to support: (1) reload reviews for Cycle 12 by Commonwealth Edison Company in accordance with 10 CFR 50.59; (2) changes resulting from analyses performed to allow equipment out-of-service; and (3) changes provided for clarification or as administrative changes. The amendment also revises the Section 3.F of the license to delete a condition requiring a Safety Evaluation for coastdown operation with abnormal feedwater temperature.

Date of issuance: January 6, 1989

Effective date: January 6, 1989 and to be implemented within 60 days.

Amendment No.: 104

Provisional Operating License No. DPR-19.: The amendment revises the Technical Specifications and the license.

Date of initial notice in **Federal Register**: October 5, 1988 (53 FR 39166). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 6, 1989.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: September 14, 1988

Brief description of amendment: This amendment provides revised Technical Specifications which incorporate new Cycle 3 reload fuel operating limits, and expands operating domains (including operation with equipment out-of-service).

Date of issuance: January 6, 1989

Effective date: January 6, 1989

Amendment No.: 41

Facility Operating License No. NPF-18. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 16, 1988 (53 FR 46140). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 6, 1989.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: December 2, 1986, and December 14, 1987.

Brief description of amendment: This amendment modified paragraph 3.F 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 this amendment.

Date of issuance: January 5, 1988

Effective date: January 5, 1988

Amendment No.: 120

Provisional Operating License No. DPR-20. The amendment revises the license.

Date of initial notice in **Federal Register:** May 4, 1988 (53 FR 15909). The Commission's related evaluation of the amendment is contained in a letter to Consumers Power Company dated January 5, 1988 and a Safeguards Evaluation Report dated January 5, 1988.

No significant hazards consideration comments received: No.

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

Dairyland Power Cooperative, Docket No. 50-409, La Crosse Boiling Water Reactor, La Crosse, Wisconsin

Date of application for amendment: December 21, 1987 as revised February 22, 1988 and October 13, 1988.

Brief description of amendment: This amendment revises the Technical Specifications (TS) for radiological environmental monitoring to delete requirements for sampling and analysis of Iodine-131. Iodine-131 is a fission product produced during reactor operations which has an 8.04 day half life and is no longer present at La Crosse since the reactor has been shut down since April 30, 1987.

Date of issuance: December 22, 1988

Effective Date: December 22, 1988

Amendment No.: 64

Possession-Only License No. DPR-45. This Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 8, 1988 (53 FR 11718). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 22, 1988.

No significant hazards consideration comments received: No.

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

Duke Power Company, et al., Docket No. 50-413, Catawba Nuclear Station, Unit 1, York County, South Carolina

Date of application for amendment: August 12, 1986, as supplemented December 14, 1987, March 1 and April 18, 1988

Brief description of amendment: The amendment substituted the standard fire protection license condition for the existing license condition.

Date of issuance: January 3, 1989

Effective date: January 3, 1989

Amendment No.: 57

Facility Operating License No. NPF-35. Amendment revised the Operating License.

Date of initial notice in **Federal Register:** November 30, 1988 (53 FR 48328). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 3, 1989.

No significant hazards consideration comments received: No

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

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

Date of application for amendments: April 15, 1988

Brief description of amendments: The amendments modified the Technical Specifications to add one containment penetration conductor overcurrent protective device to Table 3.8-1A for Unit 1 and one to Table 3.8-1B for Unit 2.

Date of issuance: January 3, 1988

Effective date: January 3, 1988

Amendment Nos.: 58 and 51

Facility Operating License Nos. NPF-35 and NPF-52. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 30, 1988 (53 FR 48329). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 3, 1988.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: October 6, 1988

Brief description of amendments: The amendments revised the Technical Specifications to reflect a modification to the pumphouse pit level instrumentation of the Nuclear Service Water System.

Date of issuance: January 10, 1989

Effective date: January 10, 1989

Amendment Nos.: 59 and 52

Facility Operating License Nos. NPF-35 and NPF-52. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 16, 1988 (53 FR 46143). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 10, 1989.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 28, 1986, as supplemented September 25 and December 23, 1986, and December 15, 1986

Brief description of amendments: The amendments extended the expiration dates for the operating licenses.

Date of issuance: December 30, 1988

Effective date: December 30, 1988

Amendment Nos.: 159 and 97

Facility Operating License Nos. DPR-57 and NPF-5. Amendments revised the Operating Licenses.

Date of initial notice in **Federal Register:** April 23, 1986 (51 FR 15397). Because the September 25 and December 23, 1986, and December 15, 1988, submittals clarified certain aspects of the original request, the substance of the changes noticed in the Federal Register and the proposed no significant hazards determination were not affected.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 30, 1988

No significant hazards consideration comments received: No.

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

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: October 25, 1988

Brief description of amendment: The amendment modified the Technical Specifications to add reference clarifications to Section 6.

Date of issuance: January 4, 1989

Effective date: January 4, 1989

Amendment No.: 15

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

Date of initial notice in **Federal Register:** November 30, 1988 (53 FR 48331). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 4, 1989.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 4, 1988

Brief description of amendment: Revises the listing of components subject to 10 CFR Part 50, Appendix J leak testing to conform to recent piping modifications made on containment penetration No. 417.

Date of Issuance: January 3, 1989

Effective date: January 3, 1989

Amendment No.: 148

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

Date of initial notice in **Federal Register:** November 30, 1988 (53 FR 48331). The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated January 3, 1989.

No significant hazards consideration comments received: No.

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

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket No. 50-498, South Texas Project, Unit 1, Matagorda County, Texas

Date of amendment request: November 7, 1988

Brief description of amendment: The amendment changed the Technical Specifications for Unit 1 to the Combined Technical Specifications for Units 1 and 2, added the positive displacement pump in a lock-out condition during cold overpressurization, added a reactor coolant pump seal isolation header pressure interlock, and made minor changes to the administrative section of the Technical Specifications.

Date of issuance: December 29, 1988.

Effective date: December 29, 1988.

Amendment No.: 4

Facility Operating License No. NPF-76. Amendment revised the Technical Specification.

Date of initial notice in **Federal Register:** November 22, 1988 (53 FR 47283). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 29, 1988.

No significant hazards consideration comments received: No.

Local Public Document Rooms Location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, Texas 77488 and Austin Public Library, 810 Guadalupe Street, Austin, Texas 78701.

Illinois Power Company, Docket No. 50-461, Clinton Power Station, Unit 1, DeWitt County Illinois

Date of application for amendment: October 30, 1987

Description of amendment request: The proposed changes will achieve consistency with a previously approved change and clarify an existing requirement.

Date of issuance: January 5, 1989

Effective date: January 5, 1989

Amendment No.: 17

Facility Operating License No. NPF-62. The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 27, 1988 (53 FR 2318). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 5, 1989.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: October 26, 1988

Brief description of amendment: Removes the offsite and facility organizational charts from the technical specifications.

Date of issuance: January 3, 1989

Effective date: January 3, 1989

Amendment No.: 108

Facility Operating License No. DPR-36: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 30, 1988 (53 FR 48333). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 3, 1989.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 18, 1988

Brief description of amendment: The change modifies the Technical Specification 4.2, "Equipment and Sampling Tests," which provides testing requirements for selected plant equipment. The proposed change modifies the specified surveillance interval for the Control Element Assembly (CEA) partial movement test in Table 4.2-2 from performance once every two weeks to monthly to agree with the Combustion Engineering standard technical specifications.

Date of issuance: January 4, 1989

Effective date: Immediately

Amendment No.: 109

Facility Operating License No. DPR-36. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 30, 1988 (53 FR 48332). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 4, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: Wiscasset Public Library, High Street, Wiscasset, Maine 04578.

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

Date of application for amendment: September 13 and September 30, 1988

Brief description of amendment: The amendment revises Technical Specification (TS) 4.4.5.1.4.a.8 to allow the inspection of steam generator tubes by insertion of the probe from the cold leg side of the steam generator tube. In addition, the amendment revises TS 3.4.6.1, "Reactor Coolant System Leakage," to decrease the allowable primary-to-secondary leakage (through any one steam generator) from 0.15 to 0.10 gpm.

Date of issuance: January 3, 1989

Effective date: January 3, 1989

Amendment No.: 138

Facility Operating License No. DPR-65. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 2, 1988 (53 FR 44253) concerning the September 13, 1988 application and November 2, 1988 (53 FR 44254) concerning the September 30, 1988 application. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 3, 1989.

No significant hazards consideration comments received: No.

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

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: July 21 and September 2, 1988

Description of amendment request: The amendment deletes the following tables, identifying electrical equipment and containment isolation valves, from the Millstone Unit 3 Technical Specifications (TS):

• Table 3.8-1 - "Containment Penetration Conductor Overcurrent Protective Devices"

• Table 3.8-2 - "Motor-Operated Valves Thermal Overload Protection Bypassed Only Under Accident Conditions"

• Table 3.8-2b - "Motor-Operated Valves Thermal Overload Protection Not Bypassed Under Accident Conditions"

• Table 3.6-2 - "Containment Isolation Valves"

The references to the above TS Tables, in their respective TS, are also deleted.

Date of issuance: December 19, 1988

Effective date: December 19, 1988

Amendment No.: 28

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

Date of initial notice in **Federal Register:** September 7, 1988 (53 FR 34609) concerning the July 21, 1988 application and October 19, 1988 (53 FR 40993) concerning the September 2, 1988 application. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 19, 1988.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 24, 1986, September 4, 1987 and November 30, 1987.

Brief description of amendments: The amendments modified paragraphs 2.C.(3) 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: Janaury 5, 1989

Effective date: Janaury 5, 1989

Amendments Nos.: 85 and 78

Facility Operating Licenses Nos. DPR-42 and DPR-60. These amendments revised the licenses.

Date of initial notice in **Federal Register:** April 6, 1988 (53 FR 11375). The Commission's related evaluation of the amendments is contained in a letter to Northern States Power Company dated Janaury 5, 1989 and a Safeguards Evaluation Report dated Janaury 5, 1989.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 11, 1988

Brief description of amendment: This amendment revised the Technical Specifications to (1) add new valves and controls to the existing list of containment isolation valves which require periodic surveillance and (2) delete Note 28 from Table 3.6.3-1 since it was no longer applicable. The amendment also deleted conditions 2.C.10 and 2.C.11 from License NPF-39 since the requirements have been fully satisfied.

Date of issuance: January 10, 1989

Effective date: January 10, 1989

Amendment No.: 13

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

Date of initial notice in **Federal Register:** October 19, 1988 (53 FR 40994). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 10, 1989.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: September 29, 1988

Brief description of amendment: The amendment changed the Technical Specifications (TSs) to (1) delete the primary containment isolation valves and instrumentation associated with the permanent removal of the reactor vessel head spray piping and (2) modify the reportability requirements for seismic monitor XR-VA-151 whenever the reactor vessel head is removed.

Date of issuance: January 11, 1989

Effective date: 60 days after date of issuance.

Amendment No.: 14

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

Date of initial notice in **Federal Register:** November 16, 1988 (53 FR 46150). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 11, 1989.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: April 14, 1988

Brief description of amendment: The amendment resolves inconsistencies between the technical specification limits and the Bases concerning maximum permissible recirculation flow imbalance when the recirculation pumps are operating at the same speed.

Date of issuance: December 28, 1988

Effective date: December 28, 1988

Amendment No.: 121

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

Date of initial notice in **Federal Register:** November 16, 1988 (53 FR 46152). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 28, 1988.

No significant hazards consideration comments received: No

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

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

Date of application for amendments: August 5, 1987 as supplemented by letter dated October 20, 1988 (TS 87-27)

Brief description of amendments: The amendments modify the Sequoyah, Units 1 and 2 Technical Specifications. The changes revise Surveillance Requirement 4.5.2.d.1 for both units. The changes reduce the setpoint, where the automatic isolation and interlock action of the residual heat removal system is verified to act, from a reactor coolant system pressure of above 750 psig to above 700 psig.

Date of issuance: December 29, 1988

Effective date: December 29, 1988

Amendment Nos.: 92, 82

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

Date of initial notice in **Federal Register:** December 30, 1987 (52 FR 49233). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 29, 1988.

No significant hazards consideration comments received: No

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

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

Date of application for amendments: June 21, 1988 (TS 88-23)

Brief description of amendments: The amendment changes the expiration date for Sequoyah Nuclear Plant (SQN), Operating License DPR-77 (Unit 1) from May 27, 2010 to September 17, 2020 and for SQN Operating License DPR-79 (Unit 2) from May 27, 2010 to September 15, 2021.

Date of issuance: December 29, 1988

Effective date: December 29, 1988

Amendment Nos.: 93, 83

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

Date of initial notice in **Federal Register:** October 5, 1988 (53 FR 39178). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 29, 1988.

No significant hazards consideration comments received: No

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

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

Date of application for amendments: November 17, 1987 (TS 87-40)

Brief description of amendments: The amendments revise the Sequoyah Nuclear Plant (SQN) Units 1 and 2 Technical Specifications. The revisions are to increase, in the conservative direction, the auxiliary feedwater (AFW) suction pressure-low trip setpoint and the allowable value of Table 3.3-4, Item 6.g, for both units for the turbine-driven AFW pump.

Date of issuance: December 29, 1988

Effective date: December 29, 1988

Amendment Nos.: 94, 84

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

Date of initial notice in **Federal Register:** February 10, 1988 (53 FR 3960). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 30, 1988.

No significant hazards consideration comments received: No

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

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

Date of application for amendment: September 30, 1988

Brief description of amendment: This amendment revised license condition 2.D.3(d) to the NA-1 Facility Operating License No. NPF-4 to state: "VEPCO may use two (2) fuel assemblies containing fuel rods clad with an advanced zirconium base alloy cladding material as described in the licensee's submittals dated February 20, 1987 and September 30, 1988." These two fuel assemblies meet the guidelines for lead test fuel assemblies and are enveloped by the existing NA-1 reload design and safety analysis limits.

NA-1 is currently operating with two (2) assemblies containing fuel rods clad with an advanced zirconium based material as approved by NRC in Amendment No. 94 for NA-1 Facility Operating License No. NPF-4 issued on May 13, 1987. The two fuel assemblies presently in place also meet the guidelines for lead test fuel assemblies and are enveloped by the existing NA-1 reload design and safety analysis limits. This amendment also granted an exemption from the requirements of 10 CFR 50.46. The evaluation of the granting of this exemption is contained in the Safety Evaluation issued with this amendment.

Date of issuance: January 3, 1989

Effective date: January 3, 1989

Amendment No.: 111

Facility Operating License No. NPF-4: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 30, 1988 (53 FR 48338). The Commission's related evaluation of the amendment is contained in an Environmental Assessment dated December 9, 1988, and in a Safety Evaluation dated January 3, 1989.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 17, 1987 (Partial)

Brief description of amendment: This amendment revised the Technical Specifications (TS) in accordance with

Virginia Electric and Power Company's Statistical DNBR Evaluation Methodology for a less restrictive negative moderator temperature coefficient for the remainder of the current NA-1 operating Cycle No. 7 only.

Date of issuance: January 3, 1989

Effective date: January 3, 1989

Amendment No.: 112

Facility Operating License No. NPF-4: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 30, 1988 (53 FR 48339). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 3, 1989.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 19, 1988

Brief description of amendment: This amendment allowed a one-time extension of 6 months for test intervals for certain surveillance tests specified in the NA-1 TS. This one-time extension is necessary to compensate for the NA-1 unscheduled steam generator tube rupture repair outage from July 15, 1987 to October 13, 1987. This unscheduled outage, together with additional time allowed for optimum fuel burnup before the next refueling outage, has resulted in a 6-month deferral of the next refueling outage for NA-1.

Date of issuance: January 9, 1989

Effective date: January 9, 1989

Amendment No.: 113

Facility Operating License No. NPF-4: Amendment revised the License.

Date of initial notice in **Federal Register:** November 16, 1988 (53 FR 46159). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 9, 1989.

No significant hazards consideration comments received: No.

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

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND FINAL DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

During the period since publication of the last biweekly notice, individual notices of issuance of amendments have been issued for the facilities as listed below. These notices were previously published as separate individual notices. They are repeated here because this biweekly notice lists all amendments that have been issued for which the Commission has made a final determination that an amendment involves no significant hazards consideration.

In this case, a prior Notice of Consideration of Issuance of Amendment and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing was issued, a hearing was requested, and the amendment was issued before any hearing because the Commission made a final determination that the amendment involves no significant hazards consideration.

Details are contained in the individual notice as cited.

Florida Power and Light Company, Docket Nos. 50-250 and 50-251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Date of application for amendments: September 21, 1988

Brief description of amendments: These amendments revised Section 3.1.2 of the Technical Specifications (TS) by incorporating modified pressure and temperature (P/T) limits for the reactor coolant system and pressurizer. The revised P/T limits are applicable up to 20 effective full power years (EFPY); the previous P/T limits expired after 10 EFPY. The amendments also revised the applicable "Bases" discussion to be consistent with the new limits, and reformatted the TS to be consistent with more recent standard TS.

Date of issuance: January 10, 1989

Effective date: January 10, 1989

Amendment Nos.: 134 and 128

Facility Operating Licenses Nos. DPR-31 and DPR-41: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** October 19, 1988 (53 FR 40988)

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

For the Nuclear Regulatory Commission

Gus C. Lainas,

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

[Doc. 89-2195 Filed 1-31-89; 8:45 am]

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