June 2, 1994

IDENTICAL LETTERS SENT TO: (See attached list of addressees)

The Honorable Joseph Lieberman, Chairman
Subcommittee on Clean Air and Nuclear Regulation
Committee on Environment
and Public Works
United States Senate
Washington, DC 20510

Dear Mr. Chairman:

Public Law 97-415, enacted on January 4, 1983, amended Section 189 of the Atomic Energy Act of 1954 to authorize the Nuclear Regulatory Commission 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.

In addition, the legislation requires the Commission to periodically (but not less frequently than once every 30 days) publish notice of any amendments issued, or proposed to be issued, under the new authority above.

Enclosed for your information is a copy of the Commission's Biweekly Notice of Applications and Amendments to Operating Licenses involving no significant hazards considerations, which was published in the Federal Register on May 25, 1994 (59 FR 27049).

Sincerely,

Original signed by

William T. Russell, Director
Office of Nuclear Reactor Regulation

Enclosure:
Federal Register
Notice

cc: Senator Alan K. Simpson

DISTRIBUTION w/o enclosure:

Central File	NRC PDR
OGC	OCA (3)
JTaylor	WTRussell/FMiraglia
SECY Mail Facility	

C+R-14
X-L-4-1, PT. 2

RETURN CONCURRENCE PAGE TO:
Dorothy Wills, 14-B-21

LA:PDII-4	D:DRPE	ADP(A)	DD:NRR	D:NRR	OCA
BClayton	SVarga	LReyes	FMiraglia	WRussell	
5/31/94	6/1/94	6/1/94	6/1/94	6/1/94	6/2/94

170054

9406210190 940602
PDR ORG NRRB
PDR

003070

DF03

The Honorable Richard H. Lehman, Chairman
Subcommittee on Energy and Mineral Resources
Committee on Natural Resources
United States House of Representatives
Washington, DC 20515

cc: Representative Barbara Vucanovich

The Honorable Philip R. Sharp, Chairman
Subcommittee on Energy and Power
Committee on Energy and Commerce
United States House of Representatives
Washington, DC 20515

cc: Representative Michael Bilirakis

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from May 2, 1994, through May 13, 1994. The last biweekly notice was published on May 12, 1994 (59 FR 24745).

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a 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 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 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.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, 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 6D22, Two White Flint North, 11555 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By June 24, 1994, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for

Domestic Licensing Proceedings" in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555 and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) The nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC 20555, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (*Project Director*): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

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

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

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

Date of amendments request: March 25, 1994.

Description of amendments request: The proposed amendment would make the following administrative changes to the Technical Specifications.

Brunswick Unit 1

1. Bases Section 2.2.1: Remove references to the Rod Sequence Control System (RSCS) in item 2 on page B 2-4.
2. Bases Section 2.2.1: Correct typographical error in acronym for hydrogen water chemistry in item 6 on page B 2-6.
3. TS 3.1.4.1: Correct typographical errors in action d, misspelling of preset, and action d.1, misspelling of BPWS acronym, on page 3/4 1-14.
4. TS Table 4.3.4-1: Remove references to the RSCS in item g of the Notes on page 3/4 3-52.
5. TS Table 3.3.5.5-1 Label each item to permit identification consistent with the scheduling system used for surveillance testing on pages 3/4 3-64a.
6. TS Table 4.3.5.5-1 Label each item to permit identification consistent with the scheduling system used for surveillance testing on page 3/4 3-64c.
7. TS 4.3.6.1.1: Correct typographical error that references Non-existent Table 4.3.6.1.1-1 to provide correct reference of Table 4.3.6.1-1 on page 3/4 3-88.
8. TS 3.4.2: Correct typographical error indicating extraneous second footnote on page 3/4 4-4.

Brunswick Unit 2

1. TS Table 2.2.1-1: Correct typographical error in item 2.b under allowable values by changing 115% to 115.5% on page 2-4.
2. Bases Section 2.2.1: Remove references to the Rod Sequence Control System (RSCS) in item 2 on page B 2-4.

3. Bases Section 2.2.1: Remove references to the Rod Sequence Control System in item 10 and revise bases description of the Select of the Select Rod Insertion consistent with removal of the RSCS on pages.

4. TS 3.1.4.1: Correct typographical error in action d.1 to correct misspelling of BPWS acronym on page 3/4 1–14.

5. TS Table 4.3.1–1: Correct grammatical omission of the word "is" in item e of the Notes on page 3/4 3–9.

6. TS Table 4.3.1–1: Remove references to the RSCS in item g of the Notes on page 3/4 3–52.

7. TS Table 3.3.5.5–1: Label each item to permit identification consistent with the scheduling system used for surveillance testing on page 3/4 3–64a.

8. TS Table 4.3.5.5–1: Label each item to permit identification consistent with the scheduling system used for surveillance testing on page 3/4 3–64c.

9. TS 3.3.6.2: Eliminate footnote, revise applicability statement and correct typographical errors in actions d and e that references non-existent Specification on page 3/4 3–93.

10. Base Section 3/4.1.4: Correct identification of Reference cited to reference 6 on page B 3/4 1–4.

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

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed change [sic] is administrative in nature. These changes do not alter the configuration or operation of the facility. The Limiting Safety Systems Settings and Safety Limits specified in the current Technical Specifications remain unchanged.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedure and emergency procedures are unaffected with the possible exception of resolving special notations that may have recognized the typographical errors that are being corrected.

3. The margins of safety are established through the Limiting Conditions of Operation, Limiting Safety Systems Settings and Safety Limits specified in the Technical Specifications. Since there are no changes to the physical design or operation of the facility, these margins will not be changed.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no 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, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Project Director: William H. Bateman

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

Date of amendment request: January 28, 1994.

Description of amendment request: The proposed amendment will remove an exception for the purge and vent valves from surveillance requirement (SR) 4.6.1.2.d and remove SR 4.6.1.2.f.

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

The proposed changes do not involve an [significant hazards consideration] SHC because the changes would not:

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

The proposed change modifies SR 4.6.1.2.d. Currently this SR indicates the purge supply and exhaust valves have an exception from the 10CFR50 Appendix J, Type B and C tests. The proposed technical specification change is consistent with current surveillance procedures and the [Final Safety Analysis Report] FSAR. The second proposed change, which removes SR 4.6.1.2.f, reflects current containment leakage surveillance requirements. The present location of SR 4.6.1.2.f could imply that containment leakage surveillance requirements are met by performing SR 4.9.9. However, SR 4.9.9 is applicable only during core alterations or movement of irradiated fuel and not during the modes when Technical Specification 3.6.1.2 is applicable. These changes have no effect on actual Appendix J testing of valves or the current plant accident analysis. Therefore, the proposed changes cannot increase the probability or consequences of an accident previously evaluated.

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

The proposed changes do not introduce any new failure modes. The plant will continue to operate as designed and there will be no change to the testing of valves. The proposed changes will not modify the plant response to the point where it can be considered a new accident. Therefore, the proposed changes will not create the possibility of a new or different kind of accident form any previously evaluated.

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

The proposed changes modify SR 4.6.1.2.d which, as presently written, indicates that the purge supply and exhaust valves are an exception to the 10CFR50 Appendix J, Type B and C test and therefore, no exception is required. This is supported by current surveillance procedures which include the purge supply and exhaust valves as part of the Type B and C tests. In addition, the proposed changes are consistent with the FSAR. FSAR Table 7.3–1 "Containment Penetrations," lists the purge supply and exhaust valves as required to receive Type B and C tests. Therefore, these proposed changes revise SR 4.6.1.2.d to reflect actual surveillance procedures and offer no revisions or reductions to current surveillance testing. Therefore, these changes will not result in a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: April 13, 1994

Description of amendment request: The proposed amendment request would revise the Technical Specifications to amend Sections 3.1.F and 4.13 to allow the repair of steam generator tubes by sleeving as an alternative to plugging. Additionally, a new tube acceptance criteria, F*, is proposed which would allow tubes that are degraded in a location not affecting structural integrity of the tube to remain in service.

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

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 would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated since the integrity of the steam generator tubes after sleeving will be equivalent to that of the original tubes. The sleeve, sleeve joint, and F* joint have been analyzed and tested for design, operating, and faulted condition loadings in accordance with NRC Regulatory [G]uide 1.121 safety factors. The potential for a tube rupture is not increased with sleeving or F*. At worst case, a tube leak would occur, resulting in a small primary to secondary leak. Primary to secondary leakage occurring from within the sleeved or F* portions of the tube is bounded by the steam generator tube rupture scenario evaluated in the Final Safety Analysis Report. In addition, the steam generator tube remains capable of performing its required heat transfer function. Placing a sleeve in the steam generator tube or leaving a tube in service with a defect in a portion of the tube that provides no function results in a more efficient steam generator than plugging an affected tube. Thus, the consequences of any accident previously evaluated are not increased because the structural integrity and the heat transfer capability of the steam generators are not significantly altered by the proposed change.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated because both the structural integrity and the heat transfer capability of the steam generators will not be significantly affected by the use of either of the sleeving processes or the implementation of the F* criteria. Testing and previous experience indicate that any primary to secondary leakage would be well below technical specification limits. In addition, in the unlikely event the defective tube failed completely at the defect, the remaining sleeve end or F* joint would restrain tube movement due to the sleeve end geometry or length of expanded contact within the tubesheet bore. Therefore, there is no threat to adjacent tubes and no other plant systems will be affected by this change. Thus, there is no potential for a new or different kind of accident.

(3) Involve a significant reduction in a margin of safety. The heat transfer capabilities of Indian Point 2 Steam Generators will be improved by utilizing the proposed sleeving process or implementing the F* criteria rather than the currently required tube plugging and subsequent loss of heat transfer area. The proposed change will allow a repaired (sleeved) tube or a tube with a tube end defect below the F* distance to remain in service, rather than completely blocking the tube's flow with plugs. Because the structural integrity of the tubes will be unaltered, the net effect of implementing the proposed change, rather than the currently required plugging procedure, will be an increase in the heat transfer characteristics of the steam generator. Westinghouse has done an evaluation of selected LOCA [loss of coolant accident] and non-LOCA transients to verify that use of sleeves resulting in a plugging equivalency at the current plant limit will not have an adverse affect on the thermal-hydraulic performance of the plant. Therefore, the margin of safety is not reduced.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: Robert A. Capra

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

Date of amendment request: November 15, 1991, as supplemented February 22, March 11, and April 7, 1994.

Description of amendment request: The amendment request, as submitted November 15, 1991, proposed completely rewritten requirements for the instrumentation and control (I&C) sections of the Palisades Technical Specifications (TS) and was initially noticed in the **Federal Register** October 28, 1992 (57 FR 48819). Since that time the licensee has updated its submittal, providing (1) changes to pages affected by intervening amendments, (2) clarifications suggested by NRC and Palisades reviewers, (3) addition of two instrument channels to the accident monitoring instruments Limiting Condition for Operation (LCO), (4) deletion of surveillance requirements for safety injection tank (SIT) instruments, as suggested by Generic Letter (GL) 93–05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation," and (5) addition of a general "Applicability" LCO which appears in the Standard TS but not in the Palisades TS. Changes (4) and (5) were not addressed in the initial proposed no significant hazards consideration (NSH) determination. The licensee's NSH analysis for these two changes was provided in its April 7, 1994, letter to the NRC and is discussed below.

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

Consumers Power Company finds that activities associated with the February 22, 1994 and March 11, 1994 Instrument and Control Technical Specification change revisions include no significant hazards, and accordingly, a no significant hazards determination in accordance with 10CFR50.92(c) is justified. The following summary supports the finding that the proposed change would not:

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

Neither the deletion of instrument surveillance requirements for the Safety Injection Tank (SIT) instrumentation nor the addition of allowance of temporarily returning inoperable equipment to service for maintenance or testing would affect the probability or consequences of an accident.

The SIT instrument channels themselves have no accident function. Their only purpose is to allow verification that the SITs themselves are operable. Surveillance requirements for these instruments were purposely deleted from STS during the Technical Specification Improvement Program. Their removal from Technical Specifications was suggested in GL 93–05.

Returning inoperable equipment to service as allowed by LCO 3.0.5 is necessary if failed channels are to be restored to operable status. The restoration of such channels enhances the ability to monitor for and mitigate abnormal operating conditions and accidents.

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

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

The proposed changes would not alter the operating conditions of the plant systems, and would not reduce the reliability of any plant safety equipment.

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

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

The proposed changes would not affect the setpoints, capacities, or operating limits for any equipment. Therefore, the proposed changes do not involve a significant reduction of a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request, as revised, involves no significant hazards consideration.

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

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

NRC Project Director: Ledyard B. Marsh.

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

Date of amendment request: April 7, 1994.

Description of amendment request: The proposed amendment would change certain Technical Specifications (TS) to relocate fuel cycle-specific parameter limits that can generally change with core reloads to a Core Operating Limits Report (COLR) in accordance with the guidance of Generic Letter 88–16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications." Several of the TS bases would also be revised to refer to limits relocated to the COLR. In each case where TS limits would be relocated to the COLR, the limits placed in the COLR would be unchanged and the appropriate bases would be revised accordingly.

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

The following evaluation supports the finding that operation of the facility in accordance with the proposed TS would not:

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

The proposed changes to the TS simply move the values and parameters for fuel cycle-specific limits from the TS to a Core Operating Limits Report (COLR). The requirements to maintain the plant within appropriate bounds are retained in the TS. The values of the cycle-specific parameter limits in the COLR are determined using an NRC-approved methodology and remain consistent with all applicable limits of the plant safety analyses that are addressed in the Final Safety Analysis Report (FSAR). A requirements for the COLR and identification of the approved methodology documents are added to the TS. There are no associated changes in plant operation. Therefore, operation of the facility in accordance with the proposed TS would not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

As discussed above, the proposed changes do not remove or alleviate any requirements to maintain the plant within the appropriate bounds. There are no associated changes in plant operation. Therefore, operation of the facility in accordance with the proposed TS would 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 proposed changes to the TS simply move the values and parameters for cycle-specific limits from the Specifications to a Core Operating Limits Report (COLR). The requirements to maintain the plant within appropriate bounds are retained in the TS. The values of the cycle-specific parameter limits in the COLR are determined using an NRC-approved methodology and remain consistent with all applicable limits of the plant safety analyses that are addressed in the Final Safety Analysis Report (FSAR). A requirement for the COLR and identification of the approved methodology documents are added to the TS. There are no associated changes in plant operation. Therefore, operation of the facility in accordance with the proposed TS would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: Ledyard B. Marsh.

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

Date of amendment request: March 29, 1994, as corrected April 26, 1994.

Date of amendment request: March 29, 1994, as corrected April 26, 1994.

Description of amendment request: The proposed amendment would modify the surveillance requirements for scram discharge volume vent and drain valves and isolation actuation instrumentation and modify the required actions and surveillance requirements for the emergency diesel generators to reduce testing during power operation. These changes are in accordance with guidance contained in Generic Letter (GL) 93–05 "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation," dated September 27, 1993.

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

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes to the frequency of testing for these components will reduce the probability of failure due to wear and eliminate the possibility of initiating transients during testing of these components. Therefore, the proposed changes will result in a decrease in the probability of previously evaluated accidents. Further, the proposed changes do not alter the design, function, or operation of the components involved and therefore, do not affect the consequences of any previously evaluated accident.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. As stated above, the proposed changes do not alter the design, function, or operation of the components involved and therefore, no new accident scenarios are created.

3. The proposed changes do not involve a significant reduction in a margin of safety. As developed in Reference 3 [NUREG–1366, "Improvement to Technical Specification Surveillance Requirements," dated December 1992] and endorsed in Reference 2 [GL 93–05], the proposed changes to the testing frequency will increase the margin of safety through reduced equipment wear and elimination of opportunities to induce transients.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: Ledyard B. Marsh.

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

Date of amendment request: April 26, 1994.

Description of amendment request: The proposed amendment would relocate tables of instrument response time limits from the Technical Specifications to the Updated Final Safety Analysis Report (UFSAR) in accordance with the guidance contained in Generic Letter 93–08 dated December 29, 1993.

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

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes delete and subsequently relocate the details of Technical Specification Table 3.3.1-2, "REACTOR PROTECTION SYSTEM RESPONSE TIMES," Table 3.3.2-3, "ISOLATION ACTUATION SYSTEM INSTRUMENTATION RESPONSE TIME," and Table 3.3.3-3, "EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES," consistent with the guidance provided by Generic Letter 93-08 dated, December 29, 1993, entitled, "Relocation of Technical Specification Tables of Instrument Response Time Limits." Generic Letter 93-08 recommends the removal and subsequent relocation of various Technical Specification tables which denote instrument and system response time limits. The response time limits and associated footnotes are proposed to be relocated to the Fermi 2 Updated Final Safety Analysis Report (UFSAR). This allows Fermi 2 to administratively control subsequent changes to the response time limit tables in accordance with 10 CFR 50.59. The procedures which contain the various response time limits are also subject to the change control provisions in the Administrative Controls section of the Technical Specifications. The proposed change only relocates the existing response time limits. The Surveillance Requirements and associated Actions are not affected and remain in the Technical Specifications. Relocating this information does not affect the initial conditions of a design basis accident or transient analysis. Since any subsequent changes to the UFSAR or procedures are evaluated in accordance with 10 CFR 50.59, no increase in the probability or consequences of an accident previously evaluated is allowed. Further, the proposed changes do not alter the design, function, or operation of the components involved and therefore, do not affect the consequences of any previously evaluated accident.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes will not impose any different operational or surveillance requirements. The changes propose to relocate these response time limit tables to other plant documents whereby adequate control of information is maintained. Further, as stated above, the proposed changes do not alter the design, function, or operation of the components involved and therefore, no new accident scenarios are created.

3. The proposed changes do not involve a significant reduction in a margin of safety. The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumption. The proposed change does not alter the scope of equipment currently required to be OPERABLE or subject to surveillance testing nor does the proposed change affect any instrument setpoints or equipment safety functions. In addition, the values to be transposed from the Technical Specifications to the UFSAR are the same as the exiting Technical Specifications. Since any future changes to these requirements in the UFSAR or procedures will be evaluated per the requirements of 10 CFR 50.59, no reduction in a margin of safety is allowed. Therefore, the change does not involve a significant reduction in margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: Ledyard B. Marsh.

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

Date of amendment request: March 30, 1994.

Description of amendment request: The proposed amendments would allow the analog channel operational test interval for radiation monitoring instrumentation to be increased from monthly to quarterly. The proposed amendments are said by the licensee to be consistent with NRC staff recommendations and guidance contained in NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements," and Generic Letter 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation."

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

Criterion 1

The requested amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated. Decreasing the frequency of the radiation monitor analog channel operational test from monthly to quarterly will have no impact upon the probability of any accident, since the radiation monitors are not accident initiating equipment. Also, no credit is taken in accident analyses for automatic actions performed by radiation monitors contained in Catawba's Technical Specifications, so the requested amendments will have no adverse impact upon the consequences of any accident.

Criterion 2

The requested amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. As stated above, the radiation monitors are not accident initiating equipment. No new failure modes can be created from an accident standpoint. The plant will not be operated in a different manner.

Criterion 3

The requested amendments will not involve a significant reduction in a margin of safety. Plant safety margins will be unaffected by the proposed changes. No safety equipment which is taken credit for in accident analyses will be affected by the requested amendments. The availability of the affected radiation monitors will be increased as a result of the proposed amendments because the monitors will not have to be made unavailable for testing as frequently. In addition, radiation monitor operating experience supports the proposed amendments. Finally, the proposed amendments are consistent with the NRC position and guidance set forth in NUREG-1366 and Generic Letter 93-05.

Based upon the preceding analyses, Duke Power Company concludes that the requested amendments do not involve a significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: David B. Matthews.

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

Date of amendment request: April 19, 1994.

Description of amendment request: The licensee proposes to change Turkey Point Units 3 and 4 Technical Specifications (TS) 4.0.5.a, "Applicability—Surveillance Requirements." The licensee proposes to delete the wording ". . . (g), except where specific written relief has been granted by the Commission pursuant to 10 CFR, Section 50.55a(g)(6)(i)" in TS 4.0.5.a, for the inservice inspection and testing programs. With the revisions to the Technical Specifications, upon finding an ASME Code requirement impractical because of prohibitive dose rates or limitations in the design, construction, or system configuration, the licensee may implement the relief request once it has been submitted to the NRC provided it has been: (1) Acceptably reviewed pursuant to 10 CFR 50.59; (2) approved by the plant staff in accordance with the administrative process described in the inservice inspection and testing programs administrative procedures, and (3) reviewed and approved by the Plant Nuclear Safety Committee.

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

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

The proposed amendments remove the wording ". . . (g), except where specific written relief has been granted by the Commission pursuant to 10 CFR, Section 50.55a(g)(6)(i)", provided a 10 CFR 50.59 evaluation is performed. The Inservice Inspection and Testing Programs are described in the Technical Specifications pursuant to 10 CFR 50.55a. In addition, the proposed amendments, in accordance with NUREG 1431 and draft NUREG 1482, provide relief to the ASME code requirement in the interim between the time of submittal of a relief request until the NRC has issued a safety evaluation and granted the relief. The changes being proposed are administrative in nature and do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. Any relief from the approved ASME Section XI code requirements will require a 10 CFR 50.59 evaluation to ensure no Technical Specification changes or unreviewed safety questions exist. Therefore, operation of the facility in accordance with the proposed amendments would not affect the probability or consequences of an accident previously analyzed.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The changes being proposed are administrative in nature and will not change the physical plant or the modes of operation defined in the Facility License. The change does not involve the addition or modification of equipment nor does it alter the design or operation of plant systems. Any reliefs from the approved ASME Section XI code requirements will require a 10 CFR 50.59 evaluation to ensure no Technical Specification changes or unreviewed safety questions exist. Therefore, operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The changes being proposed are administrative in nature and do not alter the bases for assurance that safety-related activities are performed correctly or the basis for any Technical Specification that is related to the establishment of or maintenance of a safety margin. Any reliefs from the approved ASME Section XI code requirements will require a 10 CFR 50.59 evaluation to ensure no Technical Specification changes or unreviewed safety questions exist. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199.

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

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: April 19, 1994.

Description of amendment request: The licensee proposes to change Turkey Point Units 3 and 4 Technical Specifications by increasing the surveillance interval specified for air or smoke flow test through the containment spray header from "at least once per 5 years" to "at least once per 10 years." The licensee stated that the proposed surveillance interval is consistent with both Generic Letter 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation" and NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements."

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

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

The proposed amendments extend the surveillance interval required for performing a qualitative smoke or air flow test on the containment spray headers. This surveillance test is not designed to track degradation of equipment by monitoring or trending performance. The air and smoke flow test is a test of the passive design of the containment spray nozzles, i.e., the testing demonstrates whether or not the nozzles are clogged. A single failure rendering a significant number of nozzles inoperable as a result of clogging is considered not credible. The changes being proposed do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously analyzed.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendments extend the surveillance interval required for performing a qualitative smoke or air flow test on the containment spray headers. The changes being proposed will not change the physical plant or the modes of plant operation defined in the Facility License. The change does not involve the addition or modification of equipment nor does it alter the design or operation of plant systems. Therefore, operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The revised surveillance interval proposed by this submittal will not change or otherwise influence the degree of operability assumed for the containment spray system in the plant safety analyses. The changes being proposed do not alter the bases for assurance that safety-related activities are performed correctly or the basis for any Technical Specification that is related to the establishment of or maintenance of a safety margin. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied.

Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199.

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: April 15, 1994

Description of amendment request: The proposed amendment requests the deletion of the audit program frequency requirements from Technical Specification (TS) 6.5.3 and to utilize the Operational Quality Assurance (OQA) Plan as the controlling document. This change will introduce more flexibility into audit scheduling to consider plant activities and performance. In addition, a minor editorial change has been incorporated correcting a reference in TS 6.5.1.14 in response to a finding in the Operational Safety Team Inspection report of December 23, 1993.

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

GPU Nuclear has determined that this [Technical Specification change request] TSCR poses no significant hazard as defined by the NRC in 10 CFR 50.92.

1. These changes do not affect the function of any system or component. Therefore, they do not increase the probability of occurrence or consequence of an accident previously evaluated in the [Safety Analysis Report] SAR.

2. These changes do not involve a physical change to plant configuration and they do not affect the performance of any equipment. Therefore, they do not create the possibility of an accident or malfunction of a different type than previously identified.

3. The shifting of the audit frequency requirements from the Technical Specifications to the OQA Plan and the extension of the maximum interval between audits of certain areas do not change the activities to be audited nor the scope of individual audits. Furthermore, audit frequencies are not associated with the margin of safety in the bases of any Technical Specification.

Therefore, the margin of safety is not affected by this change.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: John F. Stolz

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

Date of amendment request: April 19, 1994.

Description of amendment request: The proposed change updates and clarifies Technical Specification 3.4.B.1 to be consistent with existing Specifications 1.39 and 4.3.D (ASME Code Section XI, Article 5000 requirements).

The requested change would delete reference to the ASME Code Section XI, IS-5000 ten year hydrotest inspection interval and replace this with references to: (1) The Technical Specification 1.39 definition for Reactor Vessel Pressure Testing, and (2) the Technical Specification 3.3.A (i) Reactor Vessel Pressure Testing limits (P/T and 250 °F maximum test temperature).

The requested change will clarify that the five electromatic relief valves' (EMRV) pressure relief function may be inoperable or bypassed during system pressure testing required by ASME Code Section XI, Article IWA-5000, including system leakage and hydrostatic test, with reactor vessel completely solid, core not critical and Technical Specification 3.2.A (Core Reactivity limits) satisfied.

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

1. The requested change will not involve a significant increase in the probability or consequence of any accident previously evaluated because this change: (a) Merely updates and clarifies Technical Specification 3.4.B.1 to be consistent with other existing Technical Specifications, (b) contains no adverse changes to any existing safety function necessary for the reactor vessel solid, core not critical condition, and (c) makes no modification or physical changes to plant equipment, performance or operation necessary to respond to accidents for the reactor vessel solid, core not critical condition.

2. The requested change does not create the possibility of a new or different accident from any accident previously evaluated because this change: (a) Merely updates and clarifies Technical Specification 3.4.B.1 to be consistent with other existing Technical Specifications, (b) contains no adverse changes to any existing safety function necessary for the reactor vessel solid, core not critical condition, and (c) over pressure protection would continue to be provided by the code safety valves when the EMRV pressure relief function is bypassed.

3. A significant reduction in margin of safety is not involved because even though the EMRV pressure relief function is bypassed, over pressure protection would continue to be provided by the code safety valves. Elimination of this relief function does not affect the reactor safety analysis, since credit was not taken for the EMRV pressure relief function . . . .

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: John F. Stolz

Gulf States Utilities Company, Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: March 15, 1994.

Description of amendment request: The proposed amendment would revise the technical specifications (TS) by removing TS 3/4.3.8, "Turbine Overspeed Protection System," from the TS and relocating it to an administratively controlled document.

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

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

This change request proposes deletion of Technical Specification 3/4.3.8, "Turbine Overspeed Protection System" and relocates this requirement to an existing plant program. The purpose of overspeed protection is to minimize the possible generation of turbine fragment missiles. Excessive overspeed could potentially result in the generation of missiles which could impact and damage safety related components, equipment or structures, depending on the size and trajectory of the missiles. The proposed deletion of this specification is based on the low probability of the generation of a damaging turbine missile and other existing performance verifications of the overspeed protection system.

The turbine-generator orientation at RBS [River Bend Station] is a "favorable" orientation for reducing the probability of damage to safety-related equipment from turbine missiles since all safety-related components and structures are located in the axial direction from the turbine-generator. Turbine Overspeed Protection System is necessary for protection of the turbine from only an operational and economic point of view. The system is not essential to mitigating the consequences of an accident. The system is not used in an initial condition of a design basis accident or transient analysis. The probability of damage to safety-related equipment based on turbine manufacturer's turbine failure data was calculated to be 1.473×10^{-8} per year and is acceptably low based on the probability of turbine failure data of 4.75×10^{-7} per year as recommended by NUREG–0800. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The change proposes to relocate this requirement to an existing plant program, whereby adequate control of information is maintained. The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes to parameters governing normal plant operation. The proposed change will not impose any different operational or surveillance requirements. No new failure modes are introduced. Therefore, this proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumption. The proposed change does not alter the scope of equipment currently required to be OPERABLE or subject to surveillance testing, nor does the proposed change affect any instrument setpoints or equipment safety functions. The favorable orientation of the turbine provides a margin of safety such that the possibility of missile damage to safety-related equipment is acceptably low. Therefore the change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, D.C. 20005.

NRC Project Director: William D. Beckner.

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: April 28, 1994.

Description of amendment request: The licensee proposes to revise Technical Specification Surveillance Requirement 4.6.1.3.e to add an option which will allow the personnel airlock pneumatic system leak test to be completed in 8 hours with a pressure drop of 0.50 psi. The technical specifications currently require that the door seal pneumatic system be demonstrated operable by verifying that the system pressure does not decay more than 1.5 psi within 24 hours. The change to an 8-hour test will expedite return to power following an outage since the test is on the critical path for restart following outages.

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

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

The door pneumatic seal system pressure drop test is not altered except for providing an option to utilize a reduced test duration. A conservative acceptance criteria of 0.50 psi will be assigned to the optional short duration test thus maintaining the operability of the pneumatic seal system. The proposed change does not alter equipment or assumptions made in previously evaluated accidents, therefore the consequences of previously evaluated accidents are not increased. The probability of an accident is also unaffected because the seals are not a potential accident initiator.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

With a conservative acceptance criteria of 0.50 psi assigned to the optional 8 hour door pneumatic seal system pressure drop test the capability of the door pneumatic seal system to maintain 65 psig to the airlock seals, for a minimum of 15 days upon a loss of instrument air, is assured. Loss of plant supply air is the accident evaluated in the UFSAR [Updated Final Safety Analysis Report] section 3.8.2.1.2 and plant specification 2C268SS0006. The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

To ensure the pneumatic seal system pressure drop test is not compromised, a conservative acceptance criteria of 0.50 psi will be assigned to the 8 hour test. With the conservative acceptance criteria, the proposed change does not involve a significant reduction in the margin of safety previously evaluated.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, Texas 77488.

Attorney for licensee: Jack R. Newman, Esq., Newman & Holtzinger, P.C., 1615 L Street, N.W., Washington, D.C. 20036.

NRC Project Director: Suzanne C. Black.

North Atlantic Energy Service Corporation, Docket No. 50–443, Seabrook Station, Unit No. 1 Rockingham, New Hampshire

Date of amendment request: January 14, 1994.

Description of amendment request: The proposed amendment would change the Technical Specifications (TS) to specify the composition of the Station Operation Review Committee (SORC) based on experience and expertise vice organizational position, to implement a Station Qualified Reviewer Program (SQRP), to delete the requirement for periodic procedure reviews, to revise the time within which the Nuclear Safety Audit Review Committee (NSARC) must issue reports and minutes, and to incorporate a number of editorial changes. The editorial changes would delete certain items that are no longer applicable, would remove inconsistencies involving the names of systems and equipment and NSARC function, composition, and use of alternates, and would correct the value for the reactor coolant system volume. Other editorial changes would be made for document format consistency. The proposed amendment would affect the following TS Sections and tables: 1.31, 3.3.3.6, 3.4.1.2, 4.6.3.2, 3.7.1.2, 3/4 10.6, 5.4.2, 6.3, and 6.4, and Table 4.3–1.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below.

A. The changes do not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)).

The proposed redefinition of the composition of the SORC would not diminish the effectiveness of the SORC and would continue to ensure that the SORC has the desired experience and expertise to advise the Station Manager on all matters related to nuclear safety. The proposed change would permit operational flexibility and eliminate the need for an amendment whenever organizational changes occur. The proposed SQRP would not reduce the level of procedure review, since the SORC continues to retain responsibility to review any document requiring an evaluation pursuant to 10 CFR 50.59. The SQRP would be limited to reviewing procedures that do not affect nuclear safety.

Deleting the requirement to periodically review procedures would not diminish the review process for procedures since other programmatic requirements would continue to assure procedures are reviewed and revised when necessary.

The proposed extension of time for preparing and forwarding NSARC meeting minutes would not affect safe operation of the facility. Significant safety concerns or unreviewed safety questions would still be brought to the attention of the Senior Vice President without waiting for the release of the NSARC meeting minutes. The change would not impede in any manner prompt communication of significant concerns to the Senior Vice President. The proposed changes do not affect the manner by which the facility is operated and do not change any facility design feature or equipment. The proposed changes involve administrative or programmatic requirements or merely involve editorial changes, corrections, or clarifications. Since there is no change to the facility or operating procedures, there is no effect upon the probability or consequences of any accident previously analyzed.

B. The changes do not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because they do not affect the manner by which the facility is operated and do not change any facility design feature or equipment which affects the operational characteristics of the facility. The proposed changes involve administrative or programmatic requirements or merely involve editorial changes, corrections, or clarifications.

C. The changes do not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)) because the proposed changes do not affect the manner by which the facility is operated or involve equipment or features which affect the operational characteristics of the facility.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Exeter Public Library, 47 Front Street, Exeter, New Hampshire 03833.

Attorney for licensee: Thomas Dignan, Esquire, Ropes & Gray, One International Place, Boston, Massachusetts 02110–2624.

NRC Project Director: John F. Stolz.

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

Date of amendment request: February 10, 1992, as supplemented April 14, 1994.

Description of amendment request: The proposed amendment would remove two tables from the Technical Specifications (TS) which list reactor trip system (RTS) instrumentation response times and engineered safety features actuation system (ESFAS) instrumentation response times. These tables will be placed in the Millstone 3 Technical Requirements Manual.

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

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

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

The proposed changes to remove the RTS and ESFAS response times from the Technical Specifications will not affect the operation of the RTS and ESFAS. Operability and surveillance requirements are still maintained in the Technical Specifications and the response times will be included and maintained in the plant operating procedures. A safety evaluation and PORC [Plant Operations Review Committee] review will be required for the limits to be changed. Since the systems will not be affected by the proposed changes, there is no impact on the performance of these systems or the consequences of an accident previously analyzed.

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 the plant will continue to operate as designed, the proposed changes will not modify the plant response to the point where it can be considered a new accident.

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

The proposed changes do not have any adverse impact on the protective boundaries nor do they affect the consequences of any accident previously analyzed. The Technical Specification operability and surveillance requirements will still ensure that the systems are tested and within the limits. Changing the limits requires a safety evaluation and PORC review which will ensure that the licensing basis is maintained. Therefore, the proposed changes will not impact the margin of safety as defined in the basis of any Technical Specification.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, Connecticut 06360.

Attorney for licensee: Gerald Garfield, Esquire, Day, Berry & Howard, City Place, Hartford, Connecticut 06103–3499.

NRC Project Director: John F. Stolz.

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

Date of amendment request: April 22, 1994.

Description of amendment request: The proposed amendment would delete the requirements regarding the condenser air ejector monitor from

Tables 3.3–12 and 4.3–12 of the Millstone Unit 2 Technical Specifications.

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

The proposed technical specification change has been reviewed against the criteria of 10 CFR 50.92, and it has been determined not to involve a significant hazards consideration (SHC). Specifically, the proposed change does not:

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

Deleting the operability and surveillance requirements for the condenser air ejector monitor from Tables 3.3–12 and 4.3–12 of the Millstone Unit No. 2 Technical Specifications would leave the steam generator blowdown monitor as the primary method of monitoring and isolating steam generator blowdown. The proposed license amendment imposes stricter limitations on the operation of Millstone Unit No. 2, because it requires the use of a single monitor, the steam generator blowdown monitor, to meet the requirements of Millstone Unit No. 2 Technical Specification 3.3.3.9 (Table 3.3–12).

While NNECO [Northeast Nuclear Energy Company] is proposing to delete the operability and surveillance requirements for the condenser air ejector monitor from the Millstone Unit No. 2 Technical Specifications, there are no plans to change any of the design features or functions or the condenser air ejector monitor, or any of the specified surveillances or frequency for such surveillances. The condenser air ejector monitor will continue to isolate blowdown upon a high radiation alarm.

Additionally, steam generator blowdown isolation is required to ensure compliance with 10 CFR 20. It is not required to ensure compliance with 10 CFR 100. Therefore, the condenser air ejector monitor does not perform any safety function. The condenser air ejector monitor is not safety related. It is not credited in any radiological consequence calculations presented in the Millstone Unit No. 2 FSAR [Final Safety Analysis Report].

Based on the above, this proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed license amendment does not involve any physical changes to plant equipment or any changes to plant procedures that would be a precursor to an accident. NNECO has no plans to change any of the specified surveillances or frequency for such surveillances. The condenser air ejector monitor will continue to isolate blowdown upon a high radiation alarm. Also, the proposed license amendment imposes stricter limitations on the operation of Millstone Unit No. 2 because it requires the use of a single monitor, the steam generator blowdown monitor, to meet the requirements of Millstone Unit No. 2 Technical Specification 3.3.3.9 (Table 3.3–12). Therefore, this proposed license amendment does not create the possibility of a new or different kind of accident form any accident previously evaluated.

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

Deleting the operability and surveillance requirements for the condenser air ejector monitor from Tables 3.3–12 and 4.3–12 of the Millstone Unit No. 2 Technical Specifications would leave the steam generator blowdown monitor as the primary method of monitoring and isolating steam generator blowdown. The proposed license amendment imposes stricter limitations on the operation of Millstone Unit No. 2, because it requires the use of a single monitor, the steam generator blowdown monitor, to meet the requirements of Millstone Unit No. 2 Technical Specification 3.3.3.9 (Table 3.3–12). Therefore, this proposed license amendment does not impact or reduce the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, Connecticut 06360.

Attorney for licensee: Gerald Garfield, Esquire, Day, Berry & Howard, City Place, Hartford, Connecticut 06103–3499.

NRC Project Director: John F. Stolz.

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

Date of amendment request: April 22, 1994.

Description of amendment request: The proposed amendment would modify the Millstone Unit 2 Technical Specification Table 3.3–9 by eliminating the measurement range of 10^{-1}–10^{4} counts per second (CPS) for the entry regarding the "Wide Range Logarithmic Neutron Flux Monitor." Also the amendment would correct a few typographical and editorial errors on page V of the Index for the Millstone Unit 2 Technical Specifications.

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

NNECO [Northeast Nuclear Energy Company] has reviewed the proposed changes in accordance with 10 CFR 50.90 and has concluded that the changes do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

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

NNECO's proposal to eliminate the CPS scale for the "Wide Range Logarithmic Neutron Flux Monitor" entry in Millstone Unit No. 2 Technical Specification Table 3.3–9 will not affect the ability of Millstone Unit No. 2 to meet the intent and purpose of panel C–21's original design.

The 10^{-8}% to 100% power scale overlaps the CPS scale. The range of 10^{-8}% to 100% power for the "Wide Range Logarithmic Neutron Flux Monitor" is adequate to permit the operators to bring the unit to hot shutdown from outside the control room. Also, the instruments on C–21 are not used to provide the start-up rate signal during start-up or refueling operations. This proposed license amendment does not impact the performance of any safety-related component, system, or structure.

A review of the original design drawings concluded that this proposed change is consistent with the original plant design, and reflects the actual as-built condition of the unit. The original design drawings show that the wide range logarithmic neutron flux indicators only receive a percent power signal.

NNECO's proposals to rectify a few typographical and editorial errors on page V of the Index for the Millstone Unit No. 2 Technical Specifications are administrative in nature. They ensure that the Index accurately reflects the contents of the technical specifications.

Based on the above, the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

The proposed license amendment does not impact the performance of any safety-related component, system, or structure. Panel C–21 is required to permit the operators to bring the unit to a hot shutdown condition from a location outside the control room. Deleting the CPS range for the "Wide Range Logarithmic Neutron Flux Monitor" does not affect the ability of the operators to accomplish this function. Also, the proposed change is consistent with the original design of the plant. The proposed license amendment cannot create the possibility of a new or different kind of accident form any previously analyzed.

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

NNECO's proposal to eliminate the CPS scale for the wide range logarithmic neutron flux monitors will not affect the ability of Millstone Unit No. 2 to meet the intent and purpose of panel C–21's original design. The 10^{-8}% to 100% power scale overlaps the CPS scale. The range of 10^{-8}% to 100% power for the "Wide Range Logarithmic Neutron Flux Monitor" is adequate to permit the operators to bring the unit to hot shutdown from outside of the control room. Also, the instruments on C–21 are not used to provide the start-up rate signal during start-up or refueling operations. This proposed license amendment does not impact the performance of any safety-related component, system, or structure.

Therefore, this proposed license amendment does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, Connecticut 06360.

Attorney for licensee: Gerald Garfield, Esquire, Day, Berry & Howard, City Place, Hartford, Connecticut 06103–3499.

NRC Project Director: John F. Stolz.

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

Date of amendment request: April 25, 1994.

Description of amendment request: The proposed amendment would change the Technical Specifications concerning four related issues: (1) Power-operated relief valve (PORV) and block valve reliability; (2) low-temperature overpressure protection (LTOP); (3) boron dilution; and (4) shutdown risk management. Specifically, the proposed amendment would revise Technical Specifications 3.4.3 and 3.4.9.3 to address the issues specifically raised in Generic Letter (GL) 90–06. Technical Specifications 3.1.1.3, 3.1.2.1, 3.1.2.2, 3.1.2.3, 3.1.2.4, 3.1.2.8, 3.4.1.4, 3.4.2.1, 3.4.9.1, 3.5.3, 4.1.1.3, 4.1.2.3, 4.1.2.4, 4.4.1.4, 4.4.3.1., 4.4.3.2, 4.4.9.3.1, 4.4.9.3.2, 4.5.3.2 and 4.9.8.1 would be revised to provide consistency with the proposed changes in GL 90–06 or are related to the boron dilution issue or shutdown risk management philosophies.

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

The proposed changes do not involve an SHC [significant hazards consideration] because the changes would not:

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

The proposed changes address the operability and surveillance requirements for the charging pump, HPSI [high pressure safety injection] pumps, reactor coolant pumps, safety valves, PORVs, block valves, and the LTOP, boron dilution and SDC [shutdown cooling] systems. These changes were proposed to address four main issues: to reflect the guidance of GL 90–06 with respect to PORV and cold overpressure; to address boron dilution concerns; to address shutdown risk management lessons learned; and to address recent information on cold overpressure mitigation concerns. Generally, the changes are more restrictive than present requirements and are consistent with the recommendations of GL 90–06. Also, the changes provide the operator with additional guidance that was not previously available. Therefore, the changes will not impact the probability of occurrence or consequences of an LTOP event, boron dilution event, loss of shutdown cooling, or other event requiring emergency core cooling which has been previously analyzed.

PORV Requirements

The proposed changes to Technical Specification 3.4.3 have been made to be consistent with GL 90–06. One enhancement has been made to the guidance contained in GL 90–06 and that was to replace the phrase "because of excessive seat leakage" with the phrase "and capable of being manually cycled." Although the PORV may be designated inoperable, it may be able to be manually opened and closed and in this manner can be used to mitigate transients. For example, PORV inoperability may be due to seat leakage, instrumentation problems, automatic control problems, or other causes that do not prevent manual use and do not create a possibility for a small break LOCA. The wording changes are meant to be more specific while meeting the intent of GL 90–06. The additional enhancement to GL 90–06 includes Surveillance Requirement 4.4.3.1c whereby Millstone Unit No. 2 proposed to bench test the PORVs at a qualified laboratory under conditions representative of Mode 3 or 4 conditions. We believe this off site test will result in safer plant conditions than the in situ test proposed in the generic letter. The remaining changes to Technical Specification 3.4.3 incorporate the guidance contained in GL 90–06 and do not significantly increase the probability or consequence of an LTOP event or the failure of the PORV to operate as required.

Cold Overpressurization Protection

Changes are being proposed to Technical Specification sections 3.1.2.1, 3.1.2.3, 3.4.1.4, 3.4.2.1, 3.4.3, 3.4.9.1, 3.4.9.3, 3.5.3, 4.1.2.3, 4.4.1.4, 4.4.3.1, 4.4.3.2, 4.4.9.3.1, 4.4.9.3.2, and 4.5.3.2 to incorporate the guidance of GL 90–06 as well as enhance the availability of equipment to reduce the shutdown risk while still satisfying the cold overpressure requirements.

The proposed changes to Technical Specifications 3.1.2.1 and 3.1.2.3 will ensure only one charging pump and one HPSI pump are operable in Mode 5 or 6 with the reactor vessel head on with an available vent of less than 2.8 square inches. The remaining pumps will be secured. These proposed changes have been made to ensure Millstone Unit No. 2 does not create an LTOP condition by the operation of too many pumps injecting fluid, thereby increasing pressure in a low-temperature condition. These proposed modifications are consistent with Technical Specification 3.5.3 which has also been modified and will decrease the possibility of an LTOP condition from occurring.

The proposed change to Technical Specification 3.4.2.1 will ensure consistency between this technical specification and Technical Specification 3.4.9.3. The safety valves at Millstone Unit No. 2 are not used for LTOP mitigation. The PORVs, or RCS [reactor coolant system] vent at Millstone Unit No. 2 are used to mitigate an LTOP condition. Safety valves are required to be operable during operating conditions to automatically reduce system pressures. The use of the PORV, which allows manual control, for mitigation of an LTOP event, reduces the severity and consequence of a potential overpressure event by giving the operators more control.

The proposed changes to Technical Specification 3/4.9.3 provide enhanced operational flexibility through the use of a PORV or RCS vent. The APPLICABILITY statement has been changed for clarification purposes with no change in intent and safety implications. The ACTION requirements for the LTOP system include a 7-day allowable outage time (AOT) to restore an inoperable LTOP channel to operable status before other remedial measures would have to be taken. In addition, new Action Statement 'f' states that the provisions of Specification 3.0.4 are not applicable. Therefore, the unit may enter the Modes for which the LCO apply, during a unit shutdown or placement of the head on the reactor vessel following refueling, when an LTOP channel is inoperable. In this situation, the 7-day AOT applies for restoring the channel to operable status before other remedial measures would have to be taken. This is the same manner in which the ACTION requirements apply when an LTOP channel is determined to be inoperable while the plant is in a Mode for which the LTOP system is required to be operable.

Specifications 3.4.1.4 and 3.4.9.1 have been revised to address concerns identified in an NRC Information Notice regarding previously unconsidered pressure drops across the reactor. The modifications to these two technical specifications will ensure that unanticipated pressure rises do not occur and that there will be no increase in the probability or consequences of the LTOP event.

Based on the evaluation done in support of resolution to GL 90–06 regarding the LTOP system unavailability, NNECO concludes that additional restrictions on operation with an inoperable LTOP channel are warranted when the potential for a low-temperature overpressure event is the highest, and especially when the unit is in a water-solid condition. It is also concluded that these additional measures emphasize the importance of the LTOP system, especially while operating in a water-solid condition as the primary success path for the mitigation of overpressure transients during low-temperature operation. Therefore, these enhancements will not involve a significant increase in the probability or consequence of an accident previously evaluated.

Boron Dilution

Changes are being proposed to Technical Specifications 3.1.1.3, 3.1.2.2, 3.1.2.3, 3.1.2.4, 3.1.2.8, 4.1.1.3, 4.1.2.3, and 4.1.2.4 to provide added assurance that the boron dilution analysis remains bounding while allowing lower flow rates to reduce the potential of a loss of shutdown cooling due to vortexing at mid-loop operation.

The changes to Technical Specifications 3.1.1.3, 3.1.2.2, 3.1.2.3, 3.1.2.4, 3.1.2.8, 4.1.1.3, 4.1.2.3, 4.1.2.4, and 4.9.8.1 will not significantly increase the probability or consequences of an accident. Tagging out of a charging pump, increasing shutdown margin, and reducing SDC flow will impact results of the boron dilution accident, but will not increase the probability of initiating events.

An increase in the shutdown margin requirement as was done in Technical Specifications 3.1.2.2 and 3.1.2.8 will assure consistency with the Core Operating Limits Report which provided additional margin in a boron dilution event.

Shutdown Risk

The changes proposed to Technical Specifications 3.1.1.3, 3.1.2.1, 3.1.2.3, 3.5.3, 4.1.1.3, 4.1.2.3, 4.5.3.2 and 4.9.8.1 have been optimized to take into account shutdown risk concerns. Lower shutdown cooling flow rates are allowed to minimize the potential of a loss of shutdown cooling due to vortexing during RCS mid-loop operation.

The availability of injection sources in the shutdown modes have been optimized while still meeting the cold overpressurization requirements.

To address shutdown risk issues, the method to secure an inoperable HPSI pump has been modified. Previously, disconnecting the motor circuit breaker from its electrical power circuit was the only acceptable method of isolating this pump. Additional methods of isolating the pump have been added with the key locking of a discharge valve downstream of the HPSI pump and the tagging the valve. These actions from the control room will allow the operator the ability to quickly restore water flow and reduce the risk associated with having equipment out of service while shutdown. Inadvertent actuation is prevented by requiring the operator to obtain the key to open this discharge valve from the shift supervisor. The opening of this valve would, therefore, require the actions of two knowledgeable individuals, the operator, and the shift supervisor. The limitation on the amount of pumps available is as a direct result of LTOP concerns. This provides assurance that the LTOP requirements are met while maintaining the maximum available equipment to mitigate shutdown risk concerns.

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

The proposed changes to Technical Specifications 3.1.1.3, 3.1.2.1, 3.1.2.2, 3.1.2.4, 3.1.2.8, 3.4.1.4, 3.4.2.1, 3.4.9.1, 4.1.1.3, 4.1.2.4, and 4.9.8.1 do not create the possibility of a new or different kind of accident from any previously analyzed. The proposed changes provide clarification or additional restrictions for plant personnel concerning the operation of charging pumps, HPSI pumps, PORVs, blocking valves, and the SDC, boron dilution, and LTOP systems. The proposed technical specification changes do not introduce significant changes in the manner in which the plant is being operated. Therefore, no new failure modes are being introduced, and the potential for an unanalyzed accident is not created.

The proposed changes to Technical Specifications 3.4.3 do not create the possibility of an accident of a different type than previously evaluated, since there is no change to the design of the plant. In addition, plant operations are only being altered enough to allow a block valve and PORV to be placed in conditions which allow them to better perform their safety functions.

The proposed changes to Technical Specification 3.4.9.3 do not create the possibility of an accident of a different type than previously evaluated, since there is no change to the design of the plant and the way the plant is operated.

The proposed changes to Technical Specification 3.1.2.3 and 3.5.3 allow for the isolation of an inoperable HPSI pump by the key lock closing of a valve at the discharge of the HPSI pump and the safety tagging in the closed position. This isolation is required so that a LTOP condition does not occur. This method of isolation is required so that a LTOP condition does not occur. This method of isolation is acceptable and will not create a new or different kind of accident since it is not possible to inadvertently open this valve. A deliberate action is required by the operator, with the concurrence of the shift supervisor, to obtain the key and open the valve.

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

The proposed changes will not have an adverse impact on the protection boundaries.

With regard to the GL 90–06 modifications, there is no degradation in the operability and surveillance requirements for the PORVs and block valves and the LTOP systems. There will be no change in actual practice for, or resulting performance of, these systems. All other changes are proposed mainly to clarify each requirement. For Modes 1, 2, and 3, safety-related overpressure protection is provided by the pressurizer code safety relief valves. Therefore, there will be no adverse impact on the margin of safety as defined in the bases of any technical specification. Although any two charging pumps are allowed to be operable in a shutdown condition, the flow of these pumps is consistent with the assumptions of the boron dilution analysis. Additional pumping capability is being provided to address shutdown risk concerns, however, the limitation on pumping is tied to the vent path that is available. This will ensure that the margin of safety is not impacted.

The combined effects of reducing SDC flow, tagging out a charging pump, and increasing shutdown margin is that the required operator response times of 15 minutes in Modes 4 and 5, and 30 minutes in Mode 6 are maintained.

By reducing the allowed SDC flow rate to less than that where vortexing can occur, the potential for a loss of SDC event is being reduced. Therefore, there is no decrease in the margin of safety for the boron dilution and shutdown cooling events.

The proposed changes associated with the cold overpressure mitigation system will ensure the appropriate margin of safety is maintained by limiting RCP operation in Mode 5 and limit RCS cooldown rates. These actions will ensure an LTOP condition does not occur.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, Connecticut 06360.

Attorney for licensee: Gerald Garfield, Esquire, Day, Berry & Howard, City Place, Hartford, Connecticut 06103–3499.

NRC Project Director: John F. Stolz.

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

Date of amendment request: April 5, 1994.

Description of amendment request: This amendment will delete the frequency requirements for a number of audits listed under Technical Specification (TS) 6.5.2.8 for each unit. The proposed change also includes removing the audit requirements for the Emergency Plan and the Security Plan from the TS and relocating these requirements to each of the respective plans.

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

I. This proposal does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed Technical Specification changes to delete prescribed audit frequencies and remove the Emergency Plan and Security Plan from Technical Specifications are administrative in nature and neither directly increase or decrease the likelihood that an accident will occur. The Technical Specification changes will not impact the function or method of operation of plant systems, structures, or components. Thus, the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR is not increased by the changes. Therefore, it is concluded that the proposed changes do not increase the probability or consequences of an accident previously evaluated.

II. This proposal does not create the possibility of a new or different kind of accident or from any accident previously evaluated.

The proposed Technical Specification changes to delete prescribed audit frequencies and remove the Emergency Plan and Security Plan from Technical Specifications are administrative in nature and do not involve changes to the physical plant or operations. The proposed changes do not affect systems, structures, or components (SSCs) or the operation of these SSCs; and therefore do not create the possibility of a new or different kind of accident.

III. This change does not involve a significant reduction in a margin of safety.

The proposed Technical Specification changes to delete prescribed audit frequencies and remove the Emergency Plan and Security Plan from Technical Specifications do not involve any reductions in the margin of safety. The proposed changes will enable more effective resource utilization through performance based scheduling of audits in the affected areas. Using performance indicators and other measures of program effectiveness, potential problems can be more readily identified and audit resources can be applied to these areas to enhance performance. The proposed performance based audit process will maintain or enhance the margin of safety in the areas audited.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: Charles L. Miller.

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: March 28, 1994.

Description of amendment request: The proposed modification to Technical Specification (TS) Section 4.8.4.3.a. would increase the surveillance interval for the functional test of the Reactor Protection System (RPS). The increase would be from every six (6) months to each time the plant is in cold shutdown for a period of 24 hours, unless the test was performed in the previous six months. This change is based on guidance provided in Generic Letter 91-09, "Modification Of Surveillance Interval For The Electrical Protective Assemblies In Power Supplies For The Reactor Protection System."

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

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

The Reactor Protection System equipment subject to the proposed Technical Specifications changes are not accident initiators.

The Electrical Protective Assemblies (EPAs) specified by these proposed changes are not required to actuate in order to mitigate an accident. The functional test methodology of the RPS electrical power monitoring channels will not be effected by the proposed change in test frequency. The design and function of the EPAs will not be altered and will perform as originally designed.

A review of the RPS electrical power monitoring relays surveillance test history results was performed and supports the proposed TS changes to extend the testing interval. Fifty-one (51) surveillance tests were reviewed, and all the as-found channel calibration results were within the required TS limits. There were identified deficiencies in four (4) of the fifty-one tests performed, however, these four deficiencies did not affect the operability of the RPS EPAs. Based on good historical surveillance test results, we have concluded that the reliability of the equipment is not expected to degrade during the proposed extended test interval. Furthermore, the proposed reduced testing will result in a net decrease in the probability of occurrence of a malfunction of equipment important to safety. These malfunctions would cause an invalid inadvertent trip of the RPS which would impose unnecessary challenges on the affected unit at power. The guidance set forth in Generic Letter 91-09 states "The staff concludes that the benefit to safety of reducing the frequency of testing during power operations more than offsets the risk to safety from relaxing the surveillance requirement to test the EPAs during power operation."

Since the RPS EPAs are not accident initiators, and the design and function of the equipment will not be affected by the proposed TS changes, and the reliability of the equipment is not expected to degrade during the extended test interval, and the changes would reduce the probability of unnecessary challenges to the affected unit, we have concluded that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The design and function of the RPS EPAs will not be affected by the proposed TS changes. The failure modes of the existing equipment will remain unchanged, and no new accident types will be created. The RPS electrical power monitoring channels' functional test methodology will not be affected by the proposed change in test frequency. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Based on a review of the RPS electrical power monitoring relays surveillance test history results we have concluded that the reliability of the equipment is not expected to degrade during the proposed extended test interval. In addition, the benefit to safety by reducing the frequency of testing during power operation and the attendant possible challenges to safety systems more than offsets any risk to safety from relaxing the surveillance requirements to test the EPAs during power operation. Therefore, the proposed TS changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: J.W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101.

NRC Project Director: Charles L. Miller.

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

Date of application for amendment: May 6, 1994.

Description of amendment request: The amendment would revise Unit 1 Technical Specifications, Section 5.5.3, "Capacity," to permit an interim increase in the spent fuel storage capacity in the Unit 1 Spent Fuel Pool (SFP) from 2040 fuel assemblies to 2500 fuel assemblies.

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

1. The proposed Technical Specifications (TS) change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Increasing the spent fuel storage capacity in the Unit 1 Spent Fuel Pool (SFP) from 2040 fuel assemblies to 2500 fuel assemblies does not increase the probability of occurrence of an accident. Since all fuel handling activities will be performed using approved procedures and compatible equipment, the probability of a fuel handling accident occurring is unchanged.

Increasing the spent fuel storage capacity in the Unit 1 SFP to 2500 fuel assemblies will facilitate storing 1940 spent fuel assemblies (including contingency) that have been discharged from LGS, Units 1 and 2, and 560 low exposure fuel assemblies shipped to LGS from the Shoreham Nuclear Power Station. The decay heat load associated with the entire Shoreham fuel inventory is insignificant, since it equates to less than 5% of the heat load generated from one (1) recently discharged full power fuel bundle. Therefore, the actual decay heat load to the Unit 1 SFP will be equivalent to that which is generated from storing the 1940 spent fuel assemblies discharged from LGS, Units 1 and 2.

Increasing the spent fuel storage capacity in the Unit 1 SFP to accommodate the storage of 2500 fuel assemblies, as proposed in this TS Change Request, is bounded by the existing analysis supporting the storage of spent fuel at LGS. The existing analysis considers design inputs for structural integrity, criticality, and thermal-hydraulics and is based on the storage of 2862 spent fuel assemblies. As documented in Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup Systems," of Supplement 2 of the NRC's Safety Evaluation Report, i.e., NUREG-0991, "Safety Evaluation Report Related to the Operation of Limerick Generating Station, Units 1 and 2," the NRC indicated that based on its independent analysis the heat removal capability of the Fuel Pool Cooling and Cleanup (FPCC) system could only support [illegible]4 spent fuel assemblies. However, the LGS Unit 1 TS currently limit the storage of spent fuel to 2040 spent fuel assemblies. Since the decay heat load from the Shoreham fuel inventory (i.e., 560 fuel assemblies) is insignificant, the actual heat load to the Unit 1 SFP will be equivalent to that generated from 1940 fuel assemblies discharged from LGS, Units 1 and 2, which is less than the limit currently specified [in] the TS (i.e., 2040 fuel assemblies).

Relocating six (6) of the existing Unit 2 spent fuel storage racks to the Unit 1 SFP will be conducted in accordance with PECO Energy's Heavy Loads Program which was developed in order to implement the guidance delineated in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," such that the likelihood of a heavy load drop is precluded. The Unit 2 spent fuel storage racks are identical to those already in use in the Unit 1 SFP. Procedures will be in place to ensure that the Unit 2 spent fuel storage racks are situated in the Unit 1 SFP to insure [ensure] proper neutron poison alignment with the existing Unit 1 racks. The existing spent fuel storage racks are designed for rack-to-rack [illegible] during design basis events without [illegible] of structural integrity. The racks are als[illegible]ned to withstand the impact from a dropp[illegible] fuel assembly without the loss of structural integrity or be damaged in a way that could adversely affect the criticality analysis. Increasing the spent fuel storage capacity to accommodate the storage of 2500 spent fuel assemblies will not affect the spent fuel storage racks since the racks are specifically designed to safely store spent fuel.

This proposed TS change will not prevent the ability of the FPCC system from performing its design function to adequately cool the SFP. The FPCC system will continue to function normally and be capable of maintaining the SFP temperature at or below 140 °F. The backup cooling and makeup systems (i.e., Residual Heat Removal (RHR), Emergency Service Water (ESW), and Residual Heat Removal Service Water (RHRSW) systems) will continue to function as designed to provide an alternate source of cooling and makeup water to ensure SFP cooling is maintained. The RHR system is still capable of maintaining the SFP temperature less than 140 °F as described in LGS Updated Final Safety Analysis Report (UFSAR). Increasing the spent fuel storage capacity in the Unit 1 SFP will not increase the probability of a loss of fuel pool cooling accident or adversely affect the Refuel Floor ventilation system.

The consequences of a Fuel Handling Accident as described in the LGS UFSAR are not increased since the number of fuel assemblies stored in a SFP is not an input to the initial conditions of the accident evaluation. This accident evaluates the dropping of a spent fuel assembly and the fuel grapple assembly into the reactor core during refueling operations. A drop height of 32 feet for the spent fuel assembly and 47 feet for the fuel grapple assembly are assumed and will produce the largest number of failed fuel rods. Since the maximum possible height a fuel assembly can be dropped over the SFP does not exceed 32 feet, the consequences of a Fuel Handling Accident will not be increased by increasing the number of fuel storage cells.

The consequences of a loss of fuel pool cooling as described in Section 9.1.3.6 of the LGS UFSAR will not be increased. The event described in the UFSAR assumes that the iodine in the fuel from past refuelings is negligible, due to the long decay time. Iodine is the major contributor to thyroid dose. Since the iodine in the fuel from past refuelings is negligible, due to the long decay time, increasing the spent fuel storage capacity will not increase the dose due to the release of iodine in the SFP water resulting from boiling and therefore, the consequences are not increased.

Increasing the storage capacity in the Unit 1 SFP, on an interim basis, will not increase the probability of a malfunction of the stored spent fuel since the existing thermal-hydraulic analysis confirms that sufficient cooling capability exists to accommodate the storage of 2500 fuel assemblies in the Unit 1 SFP. As for fuel criticality, the existing analysis also confirms that the stored fuel assemblies will remain sub-critical under normal and abnormal conditions.

Increasing the storage capacity in the Unit 1 SFP will not increase the probability of a malfunction of the SFP structure or SFP liner. The existing structural analysis confirms that the SFP structure has adequate margin to prevent overstressing and meets the code requirements. Increasing the storage capacity in the Unit 1 SFP will not increase the probability of a malfunction of the spent fuel storage racks during design basis events based on the existing seismic/structural analysis.

Increasing the on-site spent fuel storage capacity will not increase the probability of a malfunction of the FPCC system. The FPCC system will continue to function as designed.

The probability of a malfunction of fuel handling equipment will not be increased since increasing the storage capacity in the Unit 1 SFP, as proposed, does not affect fuel handling equipment.

Increasing the spent fuel storage capacity does not increase the consequences of a spent fuel assembly failure since the failure of one (1) assembly will not result in additional spent fuel assembly failures.

Increasing the spent fuel storage capacity will not increase the consequences of spent fuel storage rack failure, since the existing racks have been designed/qualified to limit the consequences of a failure. A failure of, or damage to one (1) storage rack, will not result in failure or damage to another storage rack.

Increasing the spent fuel storage capacity will not increase the consequences of the failure of fuel handling equipment since the maximum expected number of fuel rods damaged by a fuel handling equipment failure remains as evaluated in the LGS UFSAR.

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

2. The proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Increasing the spent fuel storage capacity in the LGS Unit 1 SFP to permit an interim increase from 2040 fuel assemblies to 2500 fuel assemblies will not create the possibility of an accident of a different type. The Unit 1 SFP has been analyzed for criticality effects, structural effects, radiological effects, and thermal-hydraulic effects. The increase in spent fuel storage capacity will be achieved by relocating six (6) existing spent fuel storage racks from the Unit 2 SFP to the Unit 1 SFP. The spent fuel storage racks are of identical design and are passive components; therefore, the possibility of creating a new accident does not exist.

No new operating schemes or active equipment types will be required to store additional fuel bundles in the SFP. Therefore, the possibility of a different type of malfunction occurring is not created.

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

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

Since the existing TS limits for fuel handling interlocks, heavy loads restrictions, water coverage over irradiated fuel, in-core decay time, and fuel sub-criticality will be maintained, the margin of safety will not be reduced.

Therefore, the proposed TS change does not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: Charles L. Miller

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

Date of application for amendments: April 15, 1994

Description of amendment request: The proposed amendment would: (1) revise Unit 3 Technical Specification (TS) 3.3.A.2.f to correct a typographical error, (2) revise the license and TSs to change the licensee's name from Philadelphia Electric Company to PECO Energy Company, (3) revise the frequency listed in TS 4.3.A.2.a for scramming each partially or fully withdrawn operable control rod from every 24 hours to within 24 hours when operating above the rod worth minimizer low power setpoint if there are three or more inoperable control rods or if there is one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out, (4) revise TS 4.4.A.2 to allow for the replacement charge on the explosive valve for the standby liquid control system to be from either the same manufactured batch as the one fired or another batch which has been certified by having one of the batches successfully fired, (5) revise the frequency in TS 4.4.B.3 to functionally test each standby liquid control system pump loop from monthly to at least once per 92 days.

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

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed changes do not alter the operation of equipment assumed to be an initiator of any analyzed event or assumed to be available for the mitigation of accidents or transients. Proposed changes 1 and 2 are administrative in nature. Proposed change 3 to reduce the requirement to verify insertion capability from every 24 hours to a single verification when one or more control rods are stuck is sufficient to verify that the problem is not generic while providing the benefit of removing a very resource intensive requirement and permits licensed operators to focus on other, more safety significant actions. Proposed change 4 will continue to provide the necessary assurance that replacement changes on the explosive valve of the standby liquid control system will be from a batch from which a sample charge has been tested satisfactorily. Proposed change 5 modifies the allowable interval between surveillance tests for the standby liquid control system without reducing the reliability of the system while providing the benefit of reduced wear and tear on the system. Therefore, these proposed changes do not increase the probability or consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because implementation of the proposed changes do not involve any physical changes to plant systems, structures, or components. The proposed changes do not allow plant operation in any mode that is not already evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. The proposed changes do not involve a significant reduction in a margin of safety because the proposed changes do not affect the manner in which the facility is operated or change equipment or features which affect the operational characteristics of the facility. Proposed changes 1 and 2 are administrative in nature. Proposed change 3 maintains the assurance that when a scram is required that, at a minimum, the assumptions used in the accident analysis will be met. Additionally, if the initial check of control rod insertion is satisfactory, the subsequent checks are not likely to identify similar problems because operating experience shows that a [stuck] rod is rare. Once it has been determined that the same problem is not occurring in other control rods the normal surveillance frequency is sufficient to verify that scram capability is maintained. Proposed change 4 provides added flexibility for providing replacement [charges] from any batch that has had a charge successfully fired. Proposed change 4 adds flexibility while maintaining the firing reliability in excess of 99.99% for the explosive valves on the standby liquid control system. Proposed change 5 does not impact any safety analysis assumptions because the frequency of testing is not assumed in any safety analysis and standby liquid control system operability is maintained. In addition, the test frequency reduction provides reduced wear and tear on the system and increased system reliability. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105

Attorney for Licensee: J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: Charles L. Miller

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

Date of amendment request: December 20, 1989, as supplemented January 16, 1990, January 3, 1992, January 30, 1992, May 5, 1993, May 26, 1993, and March 2, 1994.

Description of amendment request: This application for an amendment to the James A. FitzPatrick Technical

Specifications proposes new Safety/Relief Valve (SRV) performance limits to take credit for the currently installed SRV capacity. Specifically, three changes to the existing SRV performance limits are proposed:

• The first permits continued plant operation with two SRVs out-ofservice. Since 7 of the 11 SRVs at FitzPatrick are also automatic depressurization system (ADS) valves, this reduces the number of ADS valves required to be operable to 5. Current specifications permit only one SRV out-of-service for 30 days.

• Secondly, the setpoints for all 11 SRVs are changed to a single nominal setpoint. Current specifications stagger the setpoints from 1090 to 1140 psig.

• The third change increases the maximum permissible setpoint tolerance from one to three percent.

The new Limiting Safety System Setting (LSSS) for reactor coolant system overpressurization protection (TS 2.2.1.B), as a result of these changes, now requires that 9 of 11 SRVs be operable at a common setpoint of 1110 psig plus or minus 3 percent.

Safety analyses were performed, using a conservative SRV setpoint of 1195 psig, which demonstrate that these proposed changes are acceptable.

Other changes, not associated with SRV performance, clarify selected portions of the Technical Specifications and correct minor typographical and editorial errors.

This "Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for Hearing" (Notice) supersedes the related Notice which was published in the Federal Register on May 15, 1990 (55 FR 20228).

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

Operation of the James A. FitzPatrick Nuclear Power Plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated. A bounding analysis (NEDC-31697P, "Updated SRV Performance Requirements for the James A. FitzPatrick Nuclear Power Plant") of the revised SRV performance requirements considered plant operation with 9 of 11 SRVs operable and with a common valve actuation pressure of 1195 psig. The analysis demonstrates that a 40 psi margin exists between the maximum anticipated pressure and the American Society of Mechanical Engineers (ASME) Code upset reactor vessel pressure limit of 1375 psig. The analyses of NEDC-31697P also demonstrate that the new SRV performance limits have no significant impact on thermal limits, ECCS/LOCA performance, HPCI/RCIC operability, containment response, containment integrity, or 10 CFR [Part] 50 Appendix R alternate shutdown capability. The analyses also considered simmer margin and downward setpoint drift.

The five miscellaneous changes clarify terminology, correct typographical errors, remove a surveillance requirement which should have been deleted as part of Amendment 130, clarify when SRV manual actuation is performed, and delete a duplicate specification. These changes are purely administrative in nature and, as such, do not impact previously evaluated accidents or equipment malfunctions.

2. Ccreate the possibility of a new or different kind of accident from those previously evaluated. The new SRV performance limits are primarily administrative changes. The only physical changes involve recalibration of SRV setpoints and operation with 2 SRVs/ADS valves out-of-service. The operation and function of the pressure relief system and [are] unaffected. No new failure modes are introduced.

The proposed miscellaneous changes are purely administrative in nature and, as such, do not create the possibility of an accident or malfunction.

3. Involve a significant reduction in the margin of safety. The new SRV performance limits slightly reduce the existing margin to vessel overpressure and the margin to the 125% mechanical overspeed trip for the HPCI and RCIC turbines. However, the reduction in the overpressure margin is insignificant (approximately 25 psi) and the plant's response to transients and accidents remains well within the limits established in General Design Criteria (GDC) 15, Standard Review Plan Section 5.2.2, and FSAR Section 4.4. The reduction in turbine overspeed margin is negligible (less than 1%), because it is within the allowable tolerance of the trip settings.

The proposed miscellaneous changes are purely administrative in nature and do not involve a reduction in safety margin.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, New York 10019.

NRC Project Director: Robert A. Capra

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

Date of amendment request: April 18, 1994.

Description of amendment request: The proposed amendment would relocate the fire protection requirements of Technical Specifications (TSs) 3.14 and 4.12, and fire brigade staffing and training requirements of TSs 6.2.2(f) and 6.4.2 from the TSs to administratively-controlled operational specifications. Specifically, the proposed changes would add the NRC standard fire protection license condition to the Operating License, update the Final Safety Analysis Report (FSAR) to include the Fire Protection Program by reference and relocate the fire protection requirements from the TSs to the Indian Point 3 Operational Specifications Manual. The proposed changes have been developed in accordance with the guidance contained in NRC Generic Letter (GL) 86-10, "Implementation of Fire Protection Requirements," and GL 88-12, "Removal of Fire Protection Requirements from the Technical Specifications."

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

Consistent with the criteria of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based on the following information:

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

Response

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

This proposed amendment merely relocates the fire protection program elements from the Technical Specifications to the Operational Specifications and the FSAR [Final Safety Analysis Report]. No reduction in content is being made to the Technical Specification requirements that are being relocated. Operating limitations will continue to be imposed, and required surveillances will continue to be performed in accordance with written procedures and instructions auditable by the NRC.

Although future proposed changes to the fire protection program elements previously located in the Technical Specifications will no longer be controlled by 10 CFR 50.91, proposed changes to the Fire Protection requirements relocated to the Operational

Specifications will be evaluated by plant administrative procedures.

Thus, programmatic controls will continue to assure that future proposed fire protection program changes will not create an unreviewed safety question.

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

Response

The possibility of an accident or malfunction of a different type than evaluated previously in the safety analysis report is not created.

This proposed amendment merely relocates the fire protection Technical Specification requirements from the Technical Specifications to the Operational Specifications. No reduction to the fire protection Technical Specification requirements is being made and thus the change does not create the possibility of a new or different accident from those previously evaluated.

As noted above, future changes to the requirements in the Operational Specifications will be evaluated by plant administrative procedures.

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

Response:

The margin of safety as defined in the bases for any technical specification is not reduced.

This proposed amendment does not involve a reduction to the approved fire protection program or Fire Protection Technical Specification requirements. The Technical Specification fire protection requirements are being relocated, with no reduction in content, to the Operational Specifications. Since there is no reduction in the requirements, there is no reduction in the margin of safety.

As noted above, proposed changes to the Fire Protection Technical Specification requirements relocated to the Operational Specifications will be evaluated by plant administrative procedures.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: April 12, 1994.

Description of amendment request: This amendment request would revise the Emergency Diesel Generator hot restart test by separating it from the 24-hour endurance run and from the load sequence testing.

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

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

The proposed changes would revise the Salem Emergency Diesel Generator (EDG) surveillance criteria to allow the hot restart test to be performed independent of the Engineered Safety Features (ESF) load sequencing test and the 24-hour endurance run. The proposed surveillance requirements would continue to demonstrate that the objectives of each of these tests are met. Specifically, the EDG's are shown to be capable of starting the ESF loads in the required sequence, operating at full load for an extended period of time, and restarting from a full load temperature condition. Therefore, the proposed changes would not adversely affect the EDG's ability to support mitigation of the consequences of any previously evaluated accident. The proposed changes to the surveillance requirements do not affect the initiation or progression of any accident sequence.

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

The proposed change affects surveillance test criteria such that increased scheduling flexibility is allowed while the test objectives associated with demonstrating EDG operability continue to be met. The proposed changes do not allow any plant configurations that are presently prohibited by the Salem Technical Specifications.

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

Surveillance testing per the proposed Technical Specifications would continue to demonstrate the ability of the EDG's to perform their intended function of providing electrical power to ESF systems needed to mitigate design basis transients, consistent with the plant safety analyses. The margin of safety demonstrated by the plant safety analyses is therefore not affected by the proposed change.

Therefore, [Public Service Electric and Gas Company] PSE&G has concluded that the changes proposed herein do not involve a Significant Hazards Consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005-3502.

NRC Project Director: Charles L. Miller.

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

Date of amendment request: April 18, 1994.

Description of amendment request: The proposed amendment will revise Sections 2.C and 2.D of the San Onofre Nuclear Generating Station, Unit 1 (SONGS 1) Operating License. Section 2.C will be revised to modify or delete several licensing conditions which either no longer apply or require revision to apply to SONGS 1 in its permanently shutdown and defueled condition. Section 2.D will be revised to exempt Fire Protection reporting from the reporting requirements of Section 2.D.

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

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

No. SONGS 1 has been permanently shut down and all fuel has been taken out of the reactor and stored in the SONGS 1 spent fuel pool. The proposed change will not modify any of the existing plant configurations, controls, procedures, or technical specification requirements necessary to assure the integrity and safe operation of the spent fuel pool.

The technical basis for deleting the four license conditions, which relate to Integrated Implementation Schedule, Cycle 11 Thermal Shield Monitoring Program, Plant Modification to Eliminate Single Failure Susceptibility of Vital Bus Automatic Transfer Function, and the NRC's Confirmatory Order of January 2, 1990, is that these license conditions were intended to assure the continued safe operation of SONGS 1 as a power producing plant. With the permanent shutdown of SONGS 1 and the issuance of its Permanently Defueled Technical Specifications (PDTS) on December 28, 1993, the plant modifications and safety programs associated with the four license conditions are no longer necessary.

The technical basis for modifying the license condition on fuel transshipment is that this license condition was intended to ensure the safety of the operating plant by putting restrictions on operation of the turbine building gantry crane. These restrictions are no longer necessary, in light of the permanent shutdown of SONGS 1.

The technical basis for modifying the license condition on physical protection is that this is necessary to update the information contained in the license condition.

The technical basis for exempting the Fire Protection Program from the reporting requirements of Section 2.D is that the applicable requirements are adequately covered in 10 CFR 50.72 and 50.73, as stated in Generic Letters 86–10 and 88–12.

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

No. No safety-related equipment will be impacted by this proposed change. Thus, there is no credible likelihood that a new or different kind of accident from any accident previously evaluated would occur as a result of this proposed change.

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

No. As explained earlier, the plant modifications and safety programs associated with the license conditions being deleted are no longer necessary. The safety-related equipment concerns that led to restrictions on operation of the turbine building gantry crane no longer exist. The modification to the license condition on physical protection will update the information contained in this license condition.

The revision to Section 2.D will make the reporting requirements regarding deficiencies in the Fire Protection Program consistent with the NRC's generic guidance on this subject.

Thus operation of the facility in accordance with this proposed change will not significantly reduce a margin of safety.

The NRC staff has reviewed the analysis of the licensee and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: James A. Beoletto, Esquire, Southern California Edison Company, P.O. Box 800, Rosemead, California 91770.

NRC Project Director: Seymour H. Weiss.

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

Date of amendment request: December 23, 1993 (TS346).

Description of amendment request: The proposed amendment would revise the BFN Units 1, 2, and 3 Technical Specifications (TS) by providing an alternate visual inspection schedule for safety-related snubbers. The licensee has stated that the amendment follows the recommendations of NRC Generic Letter (GL) 90–09, "Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions" dated December 11, 1990. GL 90–09 describes a TS line item improvement acceptable to the NRC staff. The purpose of the line item improvement is to provide a means for reducing resource demands and unnecessary occupational radiological exposure attributable to snubber inspections while continuing to provide an acceptable level of confidence in snubber operability.

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

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Implementing the guidance specified in GL 90–09 will not introduce any new failure mode and will not alter any assumptions previously made in evaluating the consequences of an accident. The proposed alternate schedule for visual inspections will maintain the same operability confidence level as the existing schedule. Also, the surveillance requirement and schedule for snubber functional testing remains the same providing a 95 percent confidence level that 90 percent to 100 percent of the snubbers operate within the specified acceptance limits. The proposed visual inspection schedule is separate from functional testing and provides additional confidence that the installed snubbers will serve their design function and are being maintained operable. The proposed changes do not affect limiting safety system settings or operating parameters, and do not modify or add any accident initiating events or parameters. Therefore, the proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Implementing the recommendations specified in GL 90–09 does not involve any physical alterations to plant equipment, changes to setpoints or operating parameters, nor does it involve any potential accident initiating event. As stated in the generic letter, the alternate schedule for snubber visual inspections maintains the same confidence level as the existing schedule. Additionally, functional testing of snubbers provides a 95 percent confidence level that 90 percent to 100 percent of the snubbers operate within specified acceptance limits. Since this TS change does not physically alter the plant equipment and the snubber confidence level remains the same there will not be any new or different accident resulting from snubber failure from any accident previously evaluated.

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

The proposed change incorporates the surveillance requirements for snubber visual inspection intervals following the guidance provided in GL 90–09. As stated in the generic letter, the proposed snubber visual inspection interval maintains the same confidence level as the existing snubber visual inspection interval. This surveillance requirement does not alter the current Limiting Condition for Operation or the accompanying actions for the snubber(s). The requirement for functional testing of safety-related snubbers is unchanged and remains the basis for the established margin of safety and assures a 95 percent confidence level that 90 percent to 100 percent of the snubbers operate within the specified acceptance limits. This functional testing along with the proposed visual inspection intervals provides adequate assurance that the snubber will perform its intended function. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11H, Knoxville, Tennessee 37902.

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: March 30, 1994.

Description of amendment request: The proposed amendment would revise the TS 3/4.1.1.1 (Reactivity Control Systems—Boration Control Systems—Boration Control—Shutdown Margin), TS 3/4.1.2.8 (Reactivity Control Systems—Borated Water Sources—Shutdown), TS 3/4.1.2.9 (Reactivity Control Systems—Borated Water Sources—Operating), Bases 3/4.1.2 (Boration Systems), TS 3.4.5.1 (Emergency Core Cooling Systems—ECCS—Core Cooling Tanks), TS 3/4.5.2

(ECCS—ECCS Subsystems), TS 3/4.5.4 (ECCS—Borated Water Storage Tank), Bases 3/4.5 (ECCS), and TS 3/4.10.4 (Special Test Exceptions—Shutdown Margin). This amendment would: (a) increase the required boration flowrate in the event the required shutdown margin is not met, (b) increase the applicable minimum boron concentration and/or volume requirements, (c) revise the applicable Action statements and Surveillance Requirements, and (d) propose several administrative and editorial changes.

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

1a. Not involve a significance increase in the probability of an accident previously evaluated because no accident initiators, conditions or assumptions are significantly affected by the proposed changes.

The proposed changes would increase the required boration flowrate in the event the required SHUTDOWN MARGIN is not met, increase the minimum required volume for the Boric Acid Addition System (BAAS) and increase the minimum required boron concentration for the Borated Water Storage Tank (BWST) and the Core Flooding Tanks (CFT). The proposed changes would also revise the Technical Specification (TS) Action Statements for the BWST and the CFT, revise the TS Surveillance Requirement relating to boron concentration sampling of the CFT, and would revise the TS Surveillance Requirements involving trisodium phosphate chemistry. In addition, various administrative and editorial changes, including changes to the TS Bases, are proposed. As stated above, none of these proposed changes involve accident initiators, conditions, or assumptions.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because no accident conditions or assumptions are affected by the proposed changes.

The proposed changes for the minimum required boron concentrations and volumes for the BAAS, BWST, and CPT comply with existing requirements to maintain a 1% delta k/k shutdown margin (SDM) at all times, and are consistent with reload and LOCA analysis. Therefore, the accident condition assumption of 1% delta k/k SDM at the initiation of an accident will still be met and the radiological consequences will be as previously evaluated.

The proposed changes do not alter the source term, containment isolation, or allowable releases. The proposed changes, therefore, will not increase the radiological consequences of a previously evaluated accident.

2a. Not create the possibility of a new kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes. As stated in 1a, the proposed changes do not affect any accident initiators and are not initiators themselves. The proposed changes do not alter any accident scenarios.

2b. Not create the possibility of a different kind of accident from any accident previously evaluated because the proposed changes only affect existing components, systems, and functions and do not introduce any new requirements that cannot be met with the existing components, systems, and functions. The proposed changes do not alter any accident scenarios.

3. Not involve a significant reduction in a margin of safety. The proposed changes to the minimum required boron concentration and volumes for the BAAS, BWST, and CFT would ensure the margin of safety for reactor subcriticality is maintained at all times for anticipated future core designs.

The proposed change to the TS Action statement to increase the required boration flowrate in the event the SHUTDOWN MARGIN requirement is not met, would ensure that the boration rate is adequate for restoring the required SHUTDOWN MARGIN for anticipated future core design.

The proposed changes to the TS Action statements for the BWST and the CFT ensure that the plant is maneuvered in a timely and conservative manner, without challenging any plant systems, while minimizing the time the plant would be exposed to a LOCA with assumptions not being met.

The proposed changes to the TS Surveillance Requirements associated with trisodium phosphate chemistry would clarify the requirements, make it easier to perform testing, minimize radwaste generation, and reduce the consequences of a potential radioactive spill. The proposed changes would also make the requirements consistent with the DBNPS Updated Safety Analysis Report.

The proposed change to the TS Surveillance Requirement associated with the boron concentration sampling of the CFT would eliminate an unnecessary requirement and make the Surveillance Requirement consistent with NUREG–1430.

None of these changes would adversely affect the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: John N. Hannon

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

Date of amendment request: April 8, 1994

Description of amendment request: The proposed amendment would revise the TS 3/4.7.1.2, Auxiliary Feedwater System, TS 3/4.7.1.7, Motor Driven Feedwater Pump System, and their applicable Bases. This amendment would: (a) Clarify the requirements for operation of the Auxiliary Feedwater System and Motor Driven Feedwater Pump System, (b) increase the surveillance intervals for testing the steam turbine driven auxiliary feedwater pumps and the electric motor driven pump, and (c) modify requirements relative to stationing an individual locally, during associated surveillance testing.

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because no change is being made to any accident initiator. The proposed changes are clarifications and the incorporations of either the recommendations of Generic Letter 93–05 or the guidance provided by NUREG–1430. Therefore, it can be concluded that the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not invalidate accident conditions or assumptions used in evaluating the radiological consequences of an accident.

2a. Not create the possibility of a new kind of accident from any accident previously evaluated because the proposed changes do not change the way the plant is operated. No new types of failures or accident initiators are introduced by the proposed changes.

2b. Not create the possibility of a different kind of accident from any accident previously evaluated because no new failure modes have been defined for any plant system or component important to safety, nor has any limiting single failure been identified as a result of the proposed changes. No different accident initiators or failure mechanisms are introduced by the proposed changes.

3. Not involve a significant reduction in a margin of safety because the proposed changes continue to ensure the availability of the Auxiliary Feedwater System and the Motor Driven Feedwater System when called

upon to perform their functions and will not adversely impact any safety analysis assumptions.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: John N. Hannon.

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

Date of amendment request: April 15, 1994.

Description of amendment request: The proposed change would revise the Technical Specifications (TS) for the North Anna Power Station, Units No. 1 and No. 2 (NA-1&2). Specifically, the proposed changes would modify the pressure/temperature operating limitations during heatup and cooldown and the Low Temperature Overpressure Protection System (LTOPS) pressure setpoints and temperatures for NA-1&2. Also, the proposed changes include revised Limiting Conditions for Operation, Action Statements, and Surveillance Requirements for the Power-Operated Relief Valves (PORVs) and block valves to address the concerns discussed in NRC Generic Letter 90-06. Additionally, the proposed changes include several editorial/administrative changes.

The NA-1&2 Reactor Coolant Systems (RCS) are protected from material failure by the imposition of restrictions on allowable pressure and temperature, and on heatup and cooldown rate. The LTOPS ensures that material integrity limits are not exceeded during the design basis overpressurization accidents. Equipment operability requirements are imposed to ensure that the assumptions of the accident analyses remain valid. The operating restrictions, setpoints, and equipment operability requirements must be revised to extend their applicability to a higher cumulative burnup, and to improve operational flexibility.

The current pressure/temperature operating limits and LTOPS setpoints are valid to 12 Effective Full Power Years (EFPY) and 17 EFPY for NA-1&2, respectively. According to the most recent estimates, the burnup applicability limits will be exceeded by NA-1 in the spring of 1996. The NA-2 pressure/temperature operating limits and LTOPS setpoints remain valid well into the year 2002. The proposed NA-1 TS include revised pressure/temperature operating limits valid to end-of-license. Although the NA-2 pressure/temperature operating limits are not being changed, the NA-2 LTOPS setpoints and associated reactor vessel integrity protection philosophy are being changed. The reactor vessel integrity protection philosophy which supports the proposed TS changes provides improved operational flexibility while maintaining an adequate margin of safety as demonstrated by the safety analysis.

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

Specifically, operation of [North Anna] Power Station in accordance with the [proposed] Technical Specification changes will not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The safety analysis demonstrates that the proposed reactor vessel protection philosophy and the associated pressure/temperature limits, LTOPS setpoints, and component operability requirements, ensure that reactor vessel integrity will be maintained during normal operation and design basis accident conditions. Specifically, adherence to the heatup/cooldown rate dependent pressure/temperature operating limits ensures that the assumed design basis flaw will not propagate during normal operation. Below the LTOPS enabling temperature, automatic actuation of the PORVs ensures that the assumed design basis flaw will not propagate under design basis low-temperature overpressurization accident conditions. Two pressurizer safety valve(s) are sufficient to relieve the overpressurization due to the inadvertent startup of two charging pumps at water solid conditions without propagation of the assumed design basis flaw. The proposed changes to address the concerns of Generic Letter 90-06 [Generic Issues 70 and 94] improve LTOPS availability and reliability by instituting requirements for PORV, block valve, and control system testing and allowed outage times for these components. Although these changes do not reduce the probability of occurrence or the consequences of the LTOPS design basis (mass and heat addition) transients, the changes provide increased assurance that pressure relieving devices will perform their design function when required.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed Technical Specifications modify pressure/temperature operating limits, LTOPS setpoints and enabling temperatures, and component operability requirements. The revised pressure/temperature operating limits, and LTOPS setpoints and enabling temperatures are only slightly different than those currently in the Technical Specifications. No operating limits or setpoints are added or deleted by the proposed changes. Therefore, it may be concluded that the operating limits and setpoint changes do not create the possibility of a new or different kind of accident. With regard to component operability requirements, restrictions on the number of charging pumps which may be operable, the number of PORVs which must be operable, and the allowable temperature difference between the steam generator primary and secondary remain unchanged. Only the setpoint temperature at which these restrictions apply have been modified. The proposed changes are entirely consistent with the reactor vessel integrity protection philosophy which ensures that the design basis reactor vessel flaw will not propagate under normal operation or postulated accident conditions. Further, the proposed changes do not invalidate . . . any component design criteria or the assumptions of any UFSAR [Updated Final Safety Analysis Report] Chapter 15 accident analyses. In addition, modifications have been made to the Technical Specifications to improve availability and reliability of PORVs and associated block valves. These changes have been made in accordance with NRC guidance in Generic Letter 90-06. It may be concluded that none of the proposed changes creates the possibility of a new or different kind of accident from any previously evaluated.

(3) involve a significant reduction in a margin of safety. As described above, the reactor vessel integrity protection philosophy ensures that the design basis assumed flaw will not propagate under normal operation or design basis accident conditions. Adherence to the Technical Specification pressure/temperature operating limits ensures that the margin to vessel fracture provided by the ASME Section XI methodology is maintained. With regard to LTOPS protection, the safety analysis demonstrates that the proposed LTOPS design ensures margins consistent with those provided by ASME Section XI Appendix G methods. This conclusion is based on industry experience with LTOPS events and engineering evaluation. Specifically, both industry experience and engineering evaluation demonstrate that LTOPS design basis events may be expected to occur at essentially isothermal conditions. Engineering evaluation demonstrates that any reduction in allowable pressure due to thermal stresses which may be expected to occur during low temperature operation is insignificant when compared to margins provided by the ASME Section XI Appendix G methods for calculating pressure/temperature operating limits. Use of the isothermal pressure/temperature limit curve as the design limit for establishing low temperature PORV set

setpoints has been approved for other utilities by the NRC. This design maximizes the operating margin above the minimum RCS pressure for reactor coolant pump (RCP) operation, thereby minimizing the probability of undesired PORV lifts during RCP startup. The proposed changes to address the concerns of Generic Letter 90–06 (Generic Issues 70 and 94) improve LTOPS availability and reliability by instituting requirements for PORV, block valve, and control system testing and allowed outage times for these components. Although these changes do not increase the margin of safety demonstrated by the analysis of the LTOPS design basis (mass and heat addition) transients, the changes provide increased assurance that pressure relieving devices will perform their design function when required.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903–2498.

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: April 19, 1994.

Description of amendment request: The proposed changes would revise the Technical Specifications (TS) for the North Anna Power Station, Units No. 1 and No. 2 (NA–1&2). Specifically, the proposed changes would modify the surveillance frequency of the control rod motion testing from monthly to quarterly in accordance with NRC Generic Letter (GL) 93–05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation" dated September 27, 1993.

The proposed changes to the surveillance requirements for the control rods at NA–1&2 are consistent with the intent of GL 93–05, which is to improve safety, decrease equipment degradation, and reduce unnecessary burden on personnel resources by reducing testing requirements that are marginal to safety.

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

Specifically, operation of North Anna Power Station in accordance with the proposed Technical Specifications changes will not:

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

The proposed change to the surveillance frequency for control rods does not increase the probability of an accident occurrence. Surveillance testing is a means of determining control rod operability and does not of itself contribute to control rod inoperability. Although reduced testing also implies a less frequent confirmation of mechanical operability, operational experience has established that the reduced testing does not decrease plant safety. Furthermore, reduced frequency testing reduces the probability of an inadvertent operational transient or misaligned control rod. There are other means available (e.g., Individual Rod Position Indicators, flux distributions anomalies) to detect a misaligned control rod. Reducing the frequency of surveillance testing will decrease the possibility of finding an inoperable control rod. Industry experience has shown that most inoperable (stuck) control rods are identified during rod drop testing and unit startup after refueling outages. Therefore, the NRC has determined that a reduced frequency surveillance test during power is acceptable to determine control rod operability (trippable).

The control rods will continue to be operated in the same manner during the surveillance testing and will be available to shutdown the reactor if a Reactor Protection System trip setpoint is reached. The operability requirements, alignment and insertion limits for the control rods remain unchanged. Since the control rods remain available (trippable) to perform their intended safety function, testing of the control rods at the proposed reduced frequency will not increase the consequences of an accident previously evaluated.

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

The proposed reduced frequency testing of the control rods does not change the way the Control Rod Drive System or the control rods are operated. The reduced frequency of testing of the control rods does not alter the operation of the Control Rod Drive System or the control rods ability to perform their intended safety function. Therefore, the reduced frequency testing of the control rods does not generate any new accident precursors. In fact, industry experience has shown that this surveillance testing may result in inadvertent reactor trips, dropped control rods, or unnecessary challenges to safety systems. Therefore, the possibility of a new or different kind of accident than previously evaluated is not created by the proposed changes in surveillance frequency of the control rods.

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

The proposed reduced frequency testing of the control rods does not change the control rod operability requirement or the way the Control Rod Drive System is operated. NUREG–1366, concluded that most stuck control rods are discovered during plant startup after refueling or during control rod drop testing. Therefore, routine surveillance testing of the control rods at the proposed reduced frequency is considered adequate to identify inoperable (stuck) control rods during operation. The reduced surveillance requirements do not affect the margin of safety in that the operability requirements remained unchanged and the existing safety analysis, which assumes the most reactive control rod sticks out of the core during accident scenarios, remains bounding. Therefore, no margins of safety are adversely affected.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903–2498.

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: April 19, 1994.

Description of amendment request: The proposed changes will modify the surveillance frequency of the control rod motion testing from monthly to quarterly.

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

Specifically, operation of Surry Power Station in accordance with the proposed Technical Specifications changes will not:

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

The proposed change to the surveillance frequency for control rods does not increase the probability of an accident occurrence. Surveillance testing is a means of determining control rod operability and does not of itself contribute to control rod inoperability. Although reduced testing also implies a less frequent confirmation of mechanical operability, operational experience has established that the reduced testing does not decrease plant safety. Furthermore, reduced frequency testing reduces the probability of an inadvertent operational transient or misaligned control rod. There are other means available (e.g., Individual Rod Position Indicators, flux distributions anomalies) to detect a misaligned control rod.

Reducing the frequency of surveillance testing will decrease the possibility of finding an inoperable control rod. Industry experience has shown that most inoperable (stuck) control rods are identified during rod drop testing and unit startup after refueling outages. Therefore, the NRC has determined that a reduced frequency surveillance test during power is acceptable to determine control rod operability (trippable).

The control rods will continue to be operated in the same manner during the surveillance testing and will be available to shutdown the reactor if a Reactor Protection System trip setpoint is reached. The operability requirements, alignment and insertion limits for the control rods remain unchanged. Since the control rods remain available (trippable) to perform their intended safety function, testing of the control rods at the proposed reduced frequency will not increase the consequences of an accident previously evaluated.

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

The proposed reduced frequency testing of the control rods does not change the way the Control Rod Drive System or the control rods are operated. The reduced frequency of testing of the control rods does not alter the operation of the Control Rod Drive System or the control rods ability to perform their intended safety function. Therefore, the reduced frequency testing of the control rods does not generate any new accident precursors. In fact, industry experience has shown that this surveillance testing may result in inadvertent reactor trips, dropped control rods, or unnecessary challenges to safety systems. Therefore, the possibility of a new or different kind of accident than previously evaluated is not created by the proposed changes in surveillance frequency of the control rods.

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

The proposed reduced frequency testing of the control rods does not change the control rod operability requirement or the way the Control Rod Drive System is operated. NUREG–1366, concluded that most stuck control rods are discovered during plant startup after refueling or during control rod drop testing. Therefore, routine surveillance testing of the control rods at the proposed reduced frequency is considered adequate to identify inoperable (stuck) control rods during operation. The reduced surveillance requirements do not affect the margin of safety in that the operability requirements remained unchanged and the existing safety analysis, which assumes the most reactive control rod sticks out of the core during accident scenarios, remains bounding. Therefore, no margins of safety are adversely affected.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Project Director: Herbert N. Berkow

Previously Published Notices of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed no Signficant Hazards Consideration Determination, and Opportunity for a Hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

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

Northeast Nuclear Energy Company, et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut Date of amendment request: April 14, 1994, as supplemented April 20, 1994.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to change the laboratory testing protocol for the charcoal absorbers for the Control Room Emergency Ventilation System (TS 3.7.6.1) and the Enclosure Building Filtration System (TS 3.6.5.1)

Date of publication of individual notice in **Federal Register**: May 4, 1994 (59 FR 23085).

Expiration date of individual notice: June 4, 1994.

Local Public Document Room location: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, Connecticut 06360.

Notice of Issuance of Amendments to Facility Operating Licenses

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, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing in connection with these actions was published in the **Federal Register** as indicated.

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

For further details with respect to the action see: (1) The applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555, and at the local public document rooms for the particular facilities involved.

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

Date of application for amendment: October 19, 1993.

Brief description of amendment: This amendment removes the low condenser vacuum scram and reduces the turbine first stage setpoint at which it is permissible to bypass the turbine control valve fast closure and the turbine stop valve closure trip (scram) signals.

Date of issuance: May 5, 1994.

Effective date: May 5, 1994.

Amendment No.: 152.

Facility Operating License No. DPR-35: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 8, 1993 (58 FR 64603). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 5, 1994.

No significant hazards consideration comments received: No.

Local Public Document Room location: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

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: April 13, 1993.

Brief Description of amendments: The amendments change the Technical Specifications to revise the design features information pertaining to the elevation at which the spent fuel storage pool is designed to prevent inadvertent draining. The amendments revise this elevation from 116 feet 4 inches to 15 feet 11 inches based on the actual spent fuel pool design.

Date of issuance: May 2, 1994.

Effective date: May 2, 1994.

Amendment Nos.: 170 and 201.

Facility Operating License Nos. DPR-71 and DPR-62. Amendments revise the Technical Specifications.

Date of initial notice in **Federal Register**: March 16, 1994 (59 FR 12359). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 2, 1994.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: May 6, 1993.

Brief description of amendments: The amendments correct an error in Technical Specification Table 3.3-2 that was made with License Amendments 128 and 110.

Date of issuance: May 11, 1994.

Effective date: May 11, 1994.

Amendment Nos.: 142 and 124.

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

Date of initial notice in **Federal Register**: August 4, 1993 (58 FR 41503). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 11, 1994.

No significant hazards consideration comments received: No.

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

Entergy Operations, Inc., Docket No. 50-382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: September 16, 1993.

Brief description of amendment: The amendment changed the Appendix A Technical Specifications for the ultimate heat sink (UHS) to clarify the requirements for the wet cooling tower fan covers, increased the test interval for starting the dry and wet tower fans from 7 days to 31 days, increased the wet bulb temperature to 80 degrees F for determining Operability, and made other editorial and clarifying changes.

Date of issuance: May 9, 1994.

Effective date: May 9, 1994.

Amendment No.: 95.

Facility Operating License No. NPF-38. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 27, 1993 (58 FR 57851). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 9, 1994.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 19, 1993, as revised March 31, 1994.

Brief description of amendments: The amendments revise surveillance requirements for station batteries based on draft IEEE Standard 450-1992, "Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Date of issuance: May 2, 1994.

Effective date: May 2, 1994.

Amendment Nos.: 71/50.

Facility Operating License Nos. NPF-68 and NPF-81: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 22, 1993 (58 FR 67847).

The March 31, 1994, letter, changed the initial request to provide increased conformance to an associated draft IEEE Standard 450 maintenance and testing practice. The revision imposes restrictions on cell replacements for degraded batteries that are in late stages of service life. These restrictions were requested by the NRC staff and do not affect the NRC staff's conclusions of no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 2, 1994.

No significant hazards consideration comments received: No.

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

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: March 14, 1994.

Brief description of amendments: The amendments change the technical specifications by adding a new Limiting Condition for Operation (LCO), 3.0.6. LCO 3.0.6 will allow equipment removed from service or declared inoperable to comply with actions to be returned to service, under administrative controls, solely to perform testing. The new LCO will provide temporary relief from the applicable action statements to perform surveillance testing required to demonstrate operability of the equipment being returned to service or the operability of other equipment.

Date of issuance: April 29, 1994.

Effective date: April 29, 1994 to be implemented within 31 days of issuance.

Amendment Nos.: Unit 1—Amendment No. 60; Unit 2—Amendment No. 49.

Facility Operating License Nos. NPF-76 and NPF-80. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 30, 1994 (58 FR 14889). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 29, 1994.

No significant hazards consideration comments received: No.

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, Texas 77488.

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

Date of application for amendment: March 24, 1993.

Brief description of amendment: The amendment revised the Technical Specifications by improving organization and clarity of Section 3.8/4.8. The amendment changes the testing requirements of the operable emergency diesel generator in Section 4.5.G.1 when the other diesel is inoperable. Also, the testing requirements of the Emergency Service Water pump and loop changed when the other pump or loop is inoperable. The amendment also makes several editorial changes.

Date of issuance: May 12, 1994.

Effective date: May 12, 1994.

Amendment No.: 197.

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

Date of initial notice in **Federal Register**: July 21, 1993 (59 FR 39051) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 12, 1994.

No significant hazards consideration comments received: No.

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

North Atlantic Energy Service Corporation, Docket No. 50-443, Seabrook Station, Unit No. 1, Rockingham, New Hampshire

Date of amendment request: September 13, 1993.

Description of amendment request: This amendment revises the Appendix A Technical Specifications relating to certain sensor errors stated in Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints. The sensor errors specified for the Power Range, Neutron Flux High Setpoint (Functional Unit 2. a.) and the Power Range, Neutron Flux Low Setpoint (Functional Unit 2. b.) are changed to incorporate the Nuclear Instrumentation System cabinet percent-full-power meter accuracy and readout error.

Date of issuance: May 9, 1994.

Effective date: As of the date of issuance, to be implemented within 60 days of issuance.

Amendment No.: 31.

Facility Operating License No. NPF-86. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 13, 1993 (58 FR 52991). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 9, 1994.

No significant hazards consideration comments received: No.

Local Public Document Room location: Exeter Public Library, 47 Front Street, Exeter, New Hampshire 03833.

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 application for amendment: March 24, 1994.

Brief description of amendment: The Technical Specifications amendment revised the plant staff requirement (specified in TS Section 6.2.2.i) to temporarily allow the Operations Manager to have held a senior reactor operator (SRO) license at a pressurized water reactor other than Indian Point 3. This temporary allowance is in effect for the period ending 3 years after restart from the 1993/1994 Performance Improvement Outage and is needed to support management changes at the facility in an effort to improve overall performance.

Date of issuance: May 3, 1994.

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 147.

Facility Operating License No. DPR-64: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 1, 1994 (59 FR 15464) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 3, 1994.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: April 28, 1993, as supplemented by letters dated August 12, 1993, November 17, 1993, February 2, 1994, and April 7, 1994.

Brief description of amendments: These amendments increase the spent fuel pool capacities for Salem, Units 1 and 2 from the current 1170 fuel assemblies to 1632 fuel assemblies. Also, the decay time for refueling operations is extended from 100 hours to 168 hours.

Date of issuance: May 4, 1994.

Effective date: May 4, 1994.

Amendment Nos. 151 and 131.

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

Date of initial notice in **Federal Register**: March 4, 1994 (59 FR 10440) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 4, 1994.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 26, 1993, as supplemented on November 30, 1993, and February 8, 1994.

Brief description of amendments: These amendments revise Technical Specifications (TS) Section 15.3.7, Section 15.4.6, and Table 15.4.1-2. The revisions incorporate items that were identified during a comparison of the accident analyses in the PBNP Safety Analysis Report (FSAR) and the Limiting Conditions for Operation and surveillance sections of the PBNP TS. The changes add systems or equipment required by the accident analyses. Testing requirements for the diesel generators are also revised to eliminate the daily testing requirement when one diesel generator is inoperable.

Date of issuance: May 11, 1994.

Effective date: May 11, 1994.

Amendment Nos.: 148 and 152.

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

Date of initial notice in **Federal Register**: August 18, 1993 (58 FR 43939) The November 30, 1993, and February 8, 1994, submittal provided additional supplemental information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 11, 1994.

No significant hazards consideration comments received: No.

Local Public Document Room location: Joseph P. Mann Library, 1516

Sixteenth Street, Two Rivers, Wisconsin 54241.

Dated at Rockville, Maryland, this 18th day of May 1994.

For the Nuclear Regulatory Commission

Steven A. Varga,

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

[FR Doc. 94-12614 Filed 5-24-94; 8:45 am]

BILLING CODE 7590-01-P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

FIRST CLASS MAIL
POSTAGE AND FEES PAID
USNRC
PERMIT NO. G-67