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 20555–0001, 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, 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 John F. Stolz: 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 20555–0001, and to J.W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the 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 dated April 25, 1996, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Dated at Rockville, Maryland, this 3rd day of May 1996.

For the Nuclear Regulatory Commission.

Frank Rinaldi,

Project Manager, Project Directorate I–2, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 96–11431 Filed 5–7–96; 8:45 am]

BILLING CODE 7590–01–P

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 April 13, 1996, through April 26, 1996. The last biweekly notice was published on April 24, 1996 (61 FR 18162).

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 20555–0001, 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, 11545 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. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By June 7, 1996, 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 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 20555–0001, 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, 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 20555–0001, 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, and at the local public document room for the particular facility involved.

Baltimore Gas and Electric Company, Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: March 28, 1996.

Description of amendments request: Pursuant to 10 CFR 50.90, the Baltimore Gas and Electric Company (BGE) hereby requests an amendment to Operating License Nos. DPR–53 and DPR–69 to reduce the moderator temperature coefficient (MTC) limit shown on Technical Specification Figure 3.1.1–1. This proposed change is necessary to support changes in the safety analyses made to accommodate a larger number of plugged steam generator (SG) tubes for future operating cycles. The proposed limit will be more restrictive than the existing limit to match the analytical assumptions. In addition, the licensee provided information to clarify the relationship of the MTC to an Anticipated Transient Without Scram event in its licensing basis.

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

The safety analyses for the current fuel cycles assume 500 tubes per steam generator (SG) are plugged and the maximum beginning-of-cycle moderator temperature coefficient (MTC) is assumed to follow the curve in Technical Specification Figure 3.1.1.-1. For the fuel cycle to be installed in Unit 1 in spring 1996, Baltimore Gas and Electric Company (BGE) assumes in the analyses that more SG tubes are plugged than the current limit, and it is necessary to credit a more restrictive (less positive) limit on the maximum positive MTC to mitigate the

Reactor Coolant System pressure and temperature increase analyzed for these events. Therefore, we are proposing a change to the allowable positive MTC limits shown on Technical Specification Figure 3.1.1–1. The proposed limit will be more restrictive than the existing limit to match the analytical assumptions. Since the safety analyses supporting an increase in the number of plugged SG tubes are applicable to both Units 1 and 2, BGE is requesting this change for both Units.

The proposed change makes the limit on the maximum positive MTC more restrictive. From an operational standpoint, a more restrictive limit on MTC will help mitigate the effect of plant transients on control of plant parameters (e.g., reactor power, pressurizer pressure, pressurizer level, etc.) Therefore, the probability of a previously analyzed accident will not be significantly increased.

The reason for the proposed change is to mitigate the effect (increased reactor coolant temperatures) of increased SG U-tube plugging on the results of the affected safety analyses. Using the more restrictive limit on the maximum positive MTC, the Loss of Load, Loss of Feedwater Flow, Feed Line Break, and Control Element Assembly Withdrawal events were reanalyzed using previously accepted methodologies. The results of these analyses are within the acceptance limits for these events. Therefore, the consequences of a previously analyzed accident will not be significantly increased.

The proposed change is similar to the examples of amendments that are considered not likely to involve significant hazards considerations given in the Statements of Consideration for 10 CFR 50.92 (51 FR 7744). The example of interest is, ''A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications, e.g., a more stringent surveillance requirement.'' The proposed change provides a more restrictive limit on the positive MTC given in Technical Specification Figure 3.1.1–1. Based on the above arguments and the similarity to an example in the Federal Register, BGE has determined that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change makes the limit on the maximum positive MTC more restrictive. The proposed change does not involve installation of new or different equipment, modify the interfaces with existing equipment, change the equipment's function, or change the method of operating the equipment. The proposed change does not affect normal plant operations or configurations. The more restrictive MTC limit will help mitigate the effect of plant transients on control of plant parameters.

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

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

The proposed change provides for a more restrictive limit for the allowable positive MTC. The more restrictive limit on the maximum positive MTC was evaluated using previously approved methodologies and compared to the existing acceptance criteria. The analyses show that the proposed change preserves the margin of safety by ensuring that the results of the safety analyses for the Loss of Load, Loss of Feedwater Flow, Feed Line Break, and Control Element Assembly Withdrawal events meet established NRC acceptance limits for these events.

In addition, this proposed change is similar to the example of amendments that are considered not likely to involve significant hazards considerations given in the Statements of Consideration for 10 CFR 50.92 (51 FR 7744). The example of interest is, ''A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications, e.g., a more stringent surveillance requirement.'' The proposed change provides a more restrictive limit on the positive MTC given in Technical Specification Figure 3.1.1–1. Based on the above arguments and the similarity to an example in the Federal Register, BGE has determined that the proposed change does not involve 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendments request involves no significant hazards consideration.

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678.

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

NRC Project Director: Susan F. Shankman, Acting.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: February 8, 1996.

Description of amendment request: The proposed amendment would remove Technical Specifications (TS) 3.3.4, Turbine Overspeed Protection; TS 3.7.12, Area Temperature Monitoring; and TS 3.11.2.6, Gas Storage Tanks; and their associated bases; and relocate them to licensee-controlled documents, such as the Final Safety Analysis Report. The licensee revised the original amendment request dated October 24, 1994, to provide supplemental information to TS 6.8.4 for administrative control program related to TS 3.11.2.6, by letters dated August 31, 1995 and February 8, 1996.

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 was previously presented in the Federal Register (59 FR 60397). The staff reviewed and determined that the proposed license amendment's revisions do not alter the original conclusion that no significant hazards considerations exist pursuant to 10 CFR 50.92.

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: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

Attorney for licensee: William D. Johnson, Vice President and Senior Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Project Director: Eugene V. Imbro.

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

Date of amendment request: December 21, 1995.

Description of amendment request: The proposed amendments would delete the requirement to place the reactor mode switch in the Shutdown position if a stuck open safety/relief valve cannot be closed within two minutes. The operator would still be required to scram the reactor if suppression pool average water temperature reaches 110 degrees Fahrenheit or greater. The licensee also proposed changes to the TS index pages to reflect Bases page changes that were accepted by the NRC staff in a letter dated May 23, 1995. Because the changes to the index pages require a license amendment, they have been included as part of this submittal.

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) Involve a significant increase in the probability or consequences of an accident previously evaluated because:

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the UFSAR. A stuck open SRV event is a mild transient which neither affects fuel limits nor radiological consequences. The two minute requirement to manually scram after a SRV becomes stuck open is not assumed or used in any transient or accident analysis in the FSAR. Removing the two minute requirement to manually scram after a SRV becomes stuck open does not change the probability of any accident evaluated in the FSAR. Removing the two minute requirement to manually scram after a SRV becomes stuck open also does not change the capability of the suppression pool during this event in case of any accident involving reactor blowdown, because the suppression pool average water temperature limit in Technical Specification 3.6.2.1 is still valid and enforced. The suppression pool average water temperature limit is the only requirement during operational conditions 1 and 2 that assures sufficient heat sink capacity in case of a LOCA in the containment. Therefore, removing the two minute requirement to manually scram after a SRV becomes stuck open would not increase the probability or consequences of any postulated accident analyzed in the FSAR.

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

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the UFSAR. This change does not effect any hardware. This is a procedural change to assure that the reactor will not be unnecessarily scrammed by the operator after a SRV is stuck open for two minutes. The reactor will still be scrammed if suppression pool average water temperature increases above 110 degrees F. Since the design basis of the suppression pool is protected by this average water temperature limit, this procedural change of removing the two minute requirement to manually scram after a SRV becomes stuck open introduces no new accident or malfunction.

(3) Involve a significant reduction in the margin of safety because:

The proposed change does not reduce the margin as defined in the bases for any Technical Specification. On the contrary, if the two minute requirement to manually scram after a SRV becomes stuck open is not removed, the operator has to scram the reactor thus challenging the RPS, the rector vessel, and other associated components, and reducing the related margin to safety. This scram would be unnecessary if the suppression pool average water temperature is below the 110 degree F limit allowed by the design basis of the suppression pool. Reactor safety or suppression pool design basis is not compromised because the suppression pool average water temperature limit alone guarantees that there would not be any 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 requested amendments involve no significant hazards consideration.

Local Public Document Room location: Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348.

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

NRC Project Director: Robert A. Capra.

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

Date of amendment request: March 4, 1996.

Description of amendment request: The proposed amendments would change the McGuire Units 1 and 2 Updated Final Safety Analysis Report to delete the seismic qualification requirement for the Containment Atmosphere Particulate Radiation Monitors.

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:

This proposed change has been evaluated against the standards in 10 CFR 50.92 and has been determined to involve no significant hazards considerations, in that operation of the facility in accordance with the proposed amendment would not:

1. [I]nvolve a significant increase in the probability or consequences of an accident previously evaluated; or

EMF38(L) is not used directly for any phase of power generation or conversion or transmission, normal decay heat removal, fuel handling, or the processing of radioactive fluids. As such, it is not an ''accident initiator''. No ''accident initiator'' is affected by the change. Thus, the probability of accidents evaluated in the FSAR is not affected by the change. It is determined that sufficient ability to determine conditions inside containment remain available for any earthquake up to and including the SSE [safe-shutdown earthquake]. Furthermore, should either EMF38(L) or EMF39(L) be found to not be functional following any earthquake, including those smaller than the OBE [Operating Basis Earthquake], the appropriate steps will be taken; i.e., declare the monitor(s) inoperable and apply the action statement for TS [technical specification] 3.4.6.1 which may require that the associated unit(s) be taken to Cold Shutdown (Mode 5) if the minimum required Reactor Coolant Leakage Detection Systems are not operable. Cold Shutdown is a mode for which neither the Emergency Core Cooling System nor the containment safeguards are required. Finally, no equipment provided to mitigate any accident is adversely affected by the change. For these reasons, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated in the SAR [safety analysis report].

2. [C]reate the possibility of a new or different type of accident from any accident previously evaluated; or

As stated above, no equipment used in direct support of power generation or conversion or transmission, normal decay heat removal, fuel handling, or processing of radioactive fluids is affected with the update. No new failure modes are identified with the change. The upper bound to an undetected leak in the Reactor Coolant System is a Loss of Coolant Accident [LOCA]. As noted above, no equipment provided to mitigate a LOCA is affected by the change. For these reasons, the change will not create a new or different type of accident from any accident previously evaluated.

3. [I]nvolve a significant reduction in a margin of safety.

It has been determined that sufficient means remain at the disposal to the operators to assess conditions within the containment following any earthquake up to and including the SSE. In particular, the ability to determine leakage with the sensitivity comparable to that of EMF38(L) can be established. This meets the intent of the Regulatory Position of RG [Regulatory Guide] 1.45. In addition, should it be determined that either EMF38(L) or EMF39(L) is not functional following any earthquake, the appropriate steps will be taken; i.e, declare the monitor(s) inoperable and apply the action statement for TS 3.4.6.1 which may require that the associated unit(s) be taken to Cold Shutdown (Mode 5) if the minimum required Reactor Coolant Leakage Detection Systems are not operable. This brings the unit(s) to a mode in which TS 3.4.6.1 does not apply. It ensures that at least the minimum required Reactor Coolant System leakage detection systems will be functional before power operations are continued following a postulated earthquake smaller than the OBE. It ensures protection of the reactor coolant pressure boundary, one of the fission product barriers. No other fission product barrier is affected by the change. Therefore, the margin of safety is not reduced.

Therefore, based on the information contained in this submittal, it is determined that no significant hazard is associated with the proposed 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: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South

Church Street, Charlotte, North Carolina 28242.

NRC Project Director: Herbert N. Berkow.

Entergy Operations, Inc., Docket Nos. 50–313 and 50–368, Arkansas Nuclear One, Unit Nos. 1 and 2, Pope County, Arkansas

Date of amendment request: April 11, 1996.

Description of amendment request: The proposed technical specification amendment modifies the reactor building leak testing requirements per Option B to 10 CFR 50, Appendix J. Option B permits performance based determination of the reactor building leak testing frequency.

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. Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The proposed changes to the Technical Specifications implement Option B of 10 CFR 50 Appendix J at ANO. The proposed changes will result in increased intervals between containment leakage tests determined through a performance based approach. The intervals between such tests are not related to conditions which cause accidents. The proposed changes do not involve a change to the plant design or operation. Therefore, this change does not involve a significant increase in the probability of any accident previously evaluated.

NUREG–1493, ''Performance-Based Containment Leak-Test Program,'' contributed to the technical bases for Option B of 10 CFR 50 Appendix J. NUREG–1493 contains a detailed evaluation of the expected leakage from containment and the associated consequences. The increased risk due to lengthening of the intervals between containment leakage tests was also evaluated and found acceptable. Using a statistical approach, NUREG–1493 determined the increase in the expected dose to the public from extending the testing frequency is extremely small. It also concluded that a small increase is justifiable due to the benefits which accrue from the interval extension. The primary benefit is in the reduction in occupational exposure. The reduction in the occupational exposure is a real reduction, while the small increase to the public is statistically derived using conservative assumptions. Therefore, this change does *not* involve a significant increase in the consequences of any accident previously evaluated.

Therefore, this change does *not* involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed change to the Technical Specifications incorporates the performance based approach authorized by Option B of 10 CFR 50 Appendix J. The interval extensions allowed by this change do not involve a change to the plant design or operation. No safety related equipment or safety functions are altered as a result of this change. The reduced testing frequency does not affect the testing methodology. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to initiation of any accidents. No new accident modes are created by extending the test intervals. Therefore, this change does *not* create the possibility of a new or different kind of accident from any previously evaluated.

3. Does Not Involve a Significant Reduction in the Margin of Safety.

The proposed change does not change the performance methodology of the containment leakage rate testing program. However, the proposed change does affect the frequency of containment leakage rate testing. With an increased frequency between tests, the proposed change does increase the probability that a increase in leakage could go undetected for a longer period of time. Operational experience has demonstrated the leak tightness of the containment buildings has been significantly below the allowable leakage limit.

The margin to safety that has the potential of being impacted by the proposed change involves the offsite dose consequences of postulated accidents which are directly related to containment leakage rates. The limitation on containment leakage rate is designed to ensure the BWN total leakage volume will not exceed the value assumed in our accident analysis. The margin to [sic] safety for the offsite dose consequences of postulated accidents directly related to containment leakage is maintained by meeting the 1.0 L. acceptance criteria. The proposed change maintains the 1.0 L. acceptance criteria.

Therefore, this change does *not* involve 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: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: William D. Beckner.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: April 11, 1996.

Description of amendment request: The proposed technical specification (TS) amendment adds low-temperature overpressure protection (LTOP) requirements to the TSs to resolve Generic Issue 94 in accordance with Generic Letter 90–06.

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. Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

This proposed change provides additional controls in the ANO–2 Technical Specification [(TS)] for ensuring that LTOP [low-temperature overpressure protection] protection is available when required. The limiting condition involving the simultaneous injection of two HPSI [high pressure safety injection] and three charging pumps to an RCS [reactor coolant system] water solid condition, was used in the calculation of the ANO–2 proposed LTOP setpoints. The methodology utilized in the LTOP setpoint analysis is based on ASME [American Society of Mechanical Engineers] Code Case N–514. The code case establishes a factor of 110 percent of the operating pressure temperature curves instead of 100 percent. The safety factor utilized by the code case provides a more reasonable vessel overpressure allowance for conditions expected under pressure loading from low temperature transients. The SITs [safety injection tanks] are required to be isolated, if not depressurized, prior to entering the LTOP enable temperature and are periodically verified to be isolated when LTOP conditions exist. The LTOP setpoint of the relief valves proposed by this technical specification [TS] change is not considered to be an initiator of any transients, but is used to mitigate an overpressure condition if such a transient were to occur.

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

2. Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The design basis event for establishing LTOP limits is the simultaneous injection of two HPSI and three charging pumps to an RCS water solid condition. The LTOP vent size of 6.38 square inches and the valve pressure setpoint of less than or equal to 430 psig are currently used for mitigation of low temperature overpressure conditions. The change in the enable setpoint was analyzed by the application of Code Case N–514 and determined to adequately ensure that this temperature [sic] setpoint will mitigate a

LTOP transient. The operator action to enable the LTOP relief valves at 220 degrees ensures that the RCS including the reactor vessel will not undergo system pressures at low temperature conditions beyond their design limits. Therefore, there will not be any impact to systems, structures or components beyond their design requirements.

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

3. Does Not Involve a Significant Reduction in the Margin of Safety.

The addition of a new specification to the ANO–2 Technical Specification [TS] will not significantly reduce the margin of safety. The LTOP safety factors are based on reanalyzed conditions for 21 effective full power years (EFPY) of operation utilizing methodology contained in ASME Code Case N–514. The LTOP evaluation under Code Case N–514 for low temperature transients is considered more appropriate than the ASME Section XI. The code case establishes a factor of 110 percent of the operating pressure temperature curves instead of 100 percent. The safety factor utilized by the code case provides a more reasonable vessel overpressure allowance for conditions expected under pressure loading from low temperature transients. Although the proposed setpoint may involve a slight reduction in a margin of safety, the enable temperature setpoint will provide an equivalent level of safety to the reactor vessel during LTOP transients and will satisfy the purpose of 10 CFR 50.60 for fracture toughness. Therefore, based on the refined methodology used to calculate ANO–2 LTOP setpoints for 21 EFPY the margin of safety will not be significantly reduced.

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: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: William D. Beckner.

Entergy Operations, Inc., et al., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: April 18, 1996.

Description of amendment request: The licensee has proposed to delete a restriction on the 24-hour emergency diesel generator operation test in Surveillance Requirement 3.8.1.14 (Page 3.8–12) of the Technical Specifications (TSs) for the Grand Gulf Nuclear Station, Unit 1. The deletion would allow the test to also be conducted during power operation (i.e., during Modes 1 and 2), instead of the current requirement to only conduct the test when the plant is shut down.

The frequency of conducting this test, the conditions of the test, and the criteria to pass the test are not being changed by this amendment request.

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 for the amendment request, which is presented below:

Entergy Operations, Inc. [(EOI)] propose[d] to change the current Grand Gulf Nuclear Station [GGNS] Technical Specifications [(TSs)]. The specific change is to modify note 2 to Surveillance 3.8.1.14. Presently, this note prohibits the performance of the 24 hour diesel maintenance run while the unit is in either Mode 1 or 2. The proposed change would remove this restriction thus allowing the 24 hour run to be performed during any mode of operation (i.e., modes 1, 2, 3, 4 or 5).

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

Entergy Operations, Inc. [EOI] has evaluated the no significant hazards consideration in its request for this license amendment and determined that no significant hazards considerations results from this change. In accordance with 10 CFR 50.91(a), Entergy Operations, Inc. [EOI] is providing the analysis of the proposed amendment against the three standards in 10 CFR 50.92(c). A description of the no significant hazards consideration determination follows:

I. The proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

The GGNS UFSAR [Updated Final Safety Analysis Report] assumes that the AC electrical power sources are designed to provide sufficient capacity, capability, redundancy and reliability to ensure that the fuel, reactor coolant system and containment design limits are not exceeded during an assumed design basis event. Specifically, the UFSAR assumes that the onsite EDG's [emergency diesel generator's] provide emergency power in the event offsite power is lost to either one or all three ESF [engineered safety feature] buses. In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the EDG's in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a design basis accident such as a LOCA.

The proposed change to permit the 24 hour testing of the EDG's during power operation does not increase the chances or consequences of any previously evaluated accident. The capability of the EDG's to supply power in a timely manner will not be compromised by permitting performance of EDG testing during periods of power operation. Design features of the EDG's and electrical systems ensures that if a LOCA [loss of coolant accident] or LOP [loss of offsite power] signal, either individually or concurrently, should occur during testing that the EDG would be returned to its ready-to-load operation (i.e., EDG running at rated speed and voltage separated from the offsite sources) or separately connected to the ESF bus providing ESF loads. As such, an EDG being tested is considered to be Operable and fully capable of meeting its intended design function. Additionally, the testing of an EDG is not a precursor to any previously evaluated accidents.

Therefore, the proposed change allowing testing of EDG's during power operation will not significantly increase the probability or consequences of an accident previously evaluated.

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

As previously discussed [above], the proposed change to permit the performance of EDG testing during power operation will not affect the operation of any system or alter any system's response to previously evaluated design basis events. The EDG's will automatically transfer from the test configuration to the ready-to-load configuration following receipt of a valid signal (i.e., LOCA or LOP). In the ready-to-load configuration, the EDG will be running at rated speed and voltage separated from the offsite source capable of automatically supplying power to the ESF buses in the event that preferred power is actually loss.

Surveillance Requirement 3.8.1.17 demonstrates that the EDG will automatically override the test mode following generation of a LOCA signal. In addition the ability of the EDG's to survive a full load reject is verified by the performance of surveillance requirement 3.8.1.10. These existing surveillance requirements along with system design features ensures that the performance of EDG testing during power operation will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, reactor coolant system and containment design limits are not exceeded. Specifically, the EDG's must be capable of automatically providing power to ESF loads in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a design basis accident in the event of a loss of preferred power.

Testing of EDG's during power operation will not affect the availability or operation of any offsite source of power. In addition, the EDG being tested remains capable of meeting its intended design functions. Therefore the proposed change to the Technical Specification Surveillance Requirement 3.8.1.14 will not result in 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: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., 12th Floor, Washington, DC 20005–3502.

NRC Project Director: William D. Beckner.

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

Date of amendment request: April 15, 1996 (TSCR No. 244).

Description of amendment request: The proposed amendment would revise Specification 5.3.1.B of the Oyster Creek Technical Specifications. The current specification prohibits handling a load greater in weight than one fuel assembly over irradiated fuel in the spent fuel storage facility. The proposed change will facilitate the off load of spent fuel to the Oyster Creek Independent Spent Fuel Storage Installation (ISFSI). Specifically, the shield plug for the dry shield canister (DSC) and the associated lifting hardware will be moved over irradiated fuel which is contained in the DSC within the transfer cask located in the Cask Drop Protection System (CDPS).

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. State the basis for the determination that the proposed activity will or will not increase the probability of occurrence or consequences of an accident.

The design features and capacity of the reactor building crane provide a significant safety factor. In addition, personnel training and other administrative controls further reduce risk. Thus, the dropping of the DSC shield plug onto a loaded DSC and causing damage to the spent fuel assemblies is not a credible event. Therefore, it does not increase the probability of or consequences of an accident.

2. State the basis for the determination that the activity does or does not create the possibility of an accident or malfunction of a different type than any previously identified in the SAR [safety analysis report].

This activity will not create the possibility of a new or different type of accident than previously evaluated in the SAR because the proposed heavy load handling exception does not create a new credible accident scenario. Dropping the shield plug on a loaded DSC and damaging spent fuel assemblies is not considered a credible event.

3. State the basis for the determination that the margin of safety is not reduced.

This activity will not involve a significant reduction in the margin of safety because the proposed heavy load handling evolution does not create a credible accident scenario.

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, NJ 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.

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

Date of amendment request: April 19, 1996.

Description of amendment request: The proposed amendment would revise Technical Specification 5.14 to add the appropriate references identifying the detailed methodology and conditions for analyzing the Small Break Loss-of-Coolant Accident (SBLOCA) to the list of the approved Core Operating Limits Report methods.

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

These Proposed Changes are administrative in nature and are consistent with the guidance set forth in the NRC Generic Letter 88–16 identifying the requirements for the inclusion of analytical methodology references in Technical Specifications as used in determining compliance with the regulatory limits.

The references, as proposed to be included in section 5.14 of the Technical Specifications, have previously been reviewed and approved by the NRC for generic applicability to PWRs [Pressurized Water Reactors]. The reports identified in the Proposed Change have been accepted by the NRC for referencing in plant licensing applications.

Since the references listed in the Proposed Change have previously been found to meet the conditions of 10 CFR 50.46 and 10 CFR Appendix K, and that the plant specific safety analysis acceptance limits have not changed or been modified, the use of these references in the analysis of SBLOCA accident for the Maine Yankee plant is consistent with prior plant specific and industry requirements and practices.

Therefore, we have concluded that the Proposed Change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the Proposed Amendment create the possibility for a new or different kind of accident?*

The Proposed Changes introduce no new mode of plant operation; do not involve the physical modification of any structure, system, or component; do not affect the function, operation or surveillance for any equipment necessary for safe operation or shutdown of the plant; and, do not involve any changes to setpoints or limits or operating parameters. The Proposed Changes are administrative in nature only.

Therefore, we have concluded that the Proposed Change cannot result in the possibility of a new or different kind of accident from that previously evaluated.

3. *Does the Proposed Amendment involve a significant reduction in a margin of safety?*

The Proposed Changes are administrative in nature, consistent with the guidance of Generic Letter 88–12, and have been reviewed previously by the NRC and found acceptable with regard to the requirements of 10 CFR 50.46 and 10 CFR Appendix K. Additionally, the plant specific safety analysis acceptance criteria has not changed from that used in the latest core reload analysis.

Therefore, we have concluded that the Proposed 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: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578.

Attorney for licensee: Mary Ann Lynch, Esquire, Maine Yankee Atomic Power Company, 329 Bath Road, Brunswick, ME 04011.

NRC Deputy Director: John A. Zwolinski.

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

Date of amendment request: February 7, 1996.

Description of amendment request: The amendment would change the operating license, the Technical Specifications, and associated Bases to permit the use of 10 CFR Part 50, Appendix J, Option B, Performance-Based Containment Leakage Rate Testing in accordance with the implementation guidance in NRC's Regulatory Guide 1.163 dated September 1995. The change to the operating license would delete, in paragraph 2.D.ii, reference to certain exemptions to Appendix J previously granted by the NRC, which would no longer be applicable once Option B is implemented.

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 operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

NMP2 [Nine Mile Point, Unit 2] is currently implementing Option A of Appendix J of 10 CFR 50 for Type A, B and C testing. The proposed change to the Operating License, the Technical Specifications and the Bases would implement Option B to Appendix J of 10 CFR 50 at NMP2 for Type A, B and C testing. Option B would allow increased testing intervals after satisfying certain performance based criteria. The proposed change also corrects an inconsistency between the restoration statements and the applicability requirements of LCO [Limiting Condition of Operation] 3.6.1.2. In addition, the proposed change affects the testing intervals for the verification of the interlocks on the primary containment air lock and for the measuring of the Hydrogen Recombiner System leakage rate.

Appendix J describes the requirements for leakage testing of the primary containment and its components penetrating the primary containment. The leakage testing interval of the primary containment and its components is not a precursor or initiator to an accident. The primary containment and its penetrations minimizes the leakage of radioactivity into the environment during an accident which pressurizes the primary containment.

The testing intervals of the air lock interlocks and of the Hydrogen Recombiner System leakage rate are also not precursors or initiators to an accident. The interlocks function to provide assurance that at least one air lock door will be closed and thereby perform its accident mitigating function of minimizing the leakage of radioactivity into the environment during accident conditions. The Hydrogen Recombiner System is manually initiated following a loss-of-coolant accident (LOCA) to maintain the hydrogen concentration within the primary containment below its flammable limit during post-LOCA conditions.

An inconsistency exists between the applicability statement of LCO 3.6.1.2 and the requirement of the restoration statements to restore prior to increasing reactor coolant system temperature over 200 °F. Eliminating this inconsistency does not diminish the requirements contained in the Technical Specifications.

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

The proposed change to the Operating License, the Technical Specifications and the Bases would replace the detailed and prescriptive technical requirements contained in Option A of Appendix J with performance based requirements and supporting regulatory/industry documents contained in Option B of Appendix J. This proposed change includes a description of the 10 CFR 50 Appendix J Testing Program Plan in Section 6.8.4.f of the Technical Specifications.

This program plan, with one exception, is consistent with RG [Regulatory Guide] 1.163. This exception to the RG is acceptable as it is technically equivalent to and replaces an exemption that was applicable to Option A of Appendix J. Therefore, this program plan establishes leakage-rate test methods, procedures, acceptance criteria and analyses which comply with Option B of Appendix J to 10 CFR 50.

The implementation of this program continues to provide adequate assurance that during a DBA [Design Basis Accident]-LOCA the primary containment and its components will continue to limit leakage rates to less than the allowable leakage rates described in the Technical Specifications and thereby limit leakage consistent with the assumptions of the accident analyses. Therefore, the increased test intervals permitted by Option B for the primary containment and its penetrations will continue to implement the safety objectives underlying the requirements of Appendix J.

As discussed under the margin of safety, the impact of the proposed change on the consequences of a release is negligible. The slight increase in the risk to the population is compensated by the corresponding risk reduction benefits associated with the reduction in component cycling, stress, and wear associated with increased test intervals.

At least one air lock door in each air lock will continue to be closed during the onset of an accident that would release radioactivity into primary containment. Therefore, the air lock interlocks continue to provide assurance that at least one leak tested barrier will limit leakage during accident conditions.

The Hydrogen Recombiner System will continue to operate to maintain the hydrogen concentration within the primary containment below its flammable limit during post-LOCA conditions. This provides assurance that primary containment integrity will not be challenged by hydrogen burns.

Eliminating the inconsistency between the restoration statements and the applicability requirements of LCO 3.6.1.2 does not diminish the requirements contained in the Technical Specifications. The Technical Specifications continue to require that the leakage limits of LCO 3.6.1.2 be met prior to entering OPERATIONAL CONDITIONS 1, 2, or 3 (i.e., temperature greater than 200 °F).

Accordingly, operation with the proposed change to the Operating License, the Technical Specifications and the Bases will not significantly increase the consequences of an accident previously evaluated.

2. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change would implement Option B of Appendix J of 10 CFR 50 for Type A, B and C testing. Option B would allow increased testing intervals after satisfying certain performance based criteria. The proposed change also corrects any inconsistency between the restoration statements and the applicability requirements of LCO 3.6.1.2. In addition, the proposed change affects the testing intervals for the interlocks on the primary containment air lock and for the measuring of the Hydrogen Recombiner System leakage rate.

No new plant operating modes, system operating configurations nor failure modes are introduced by the proposed change. The primary containment and its penetrations will continue to perform their accident mitigating function. The Hydrogen Recombiner System will continue to function to prevent hydrogen burns within primary containment during post-LOCA conditions.

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

3. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

A regulatory impact analysis of implementing performance-based requirements indicates that relaxing the frequency of Type A, B and C testing leads to an increase in overall reactor risk of approximately two percent. As indicated in the Staff's Regulatory Impact Analysis, this increase is considered to be marginal to safety.

As indicated above, increasing test intervals can slightly increase the risk to the population associated with the consequences of a release; however, this is compensated by the corresponding risk reduction benefits associated with the reduction in component cycling, stress, and wear associated with increased test intervals. Therefore, when considering the total integrated risk, the risk associated with increased test intervals is negligible.

The proposed change is consistent with current plant safety analyses. In addition, the proposed change does not require revisions to the design of NMP2. As such, the proposed individual changes will maintain the same level of reliability of the equipment associated with containment integrity, assumed to operate in the plant safety analysis, or provide continued assurance that specified parameters affecting leak rate integrity, will remain within their acceptance limits.

The as-left leakage after performing a required leakage test continues to be less than 0.60 La for combined Type B and C leakage and less than or equal to 0.75 La for Type A leakage. These as-left acceptance criteria and the testing frequency as established by the 10 CFR 50 Appendix J Testing Program Plan provide assurance that the measured leakage rate will not exceed the maximum allowable leakage of La during plant operation.

Visual examination of accessible interior and exterior surfaces of the primary containment continues to be performed prior to initiating a Type A test. The total number of visual examinations performed will continue to be three times during a 10-year period. Therefore, visual examinations of the primary containment will continue to allow for the timely uncovering of evidence of structural deterioration and satisfy the requirements of RG 1.163.

The primary containment air lock interlocks will be tested prior to conducting an air lock seal leakage test. This testing requirement continues to provide adequate assurance that at least one leak tested air lock door in each air lock will be closed during accident conditions.

The measuring of the Hydrogen Recombiner System Leakage rate will continue to be included as part of the overall integrated leakage rate test. The test schedule for measuring system leakage will also continue to coincide with the schedule for performing a Type A test.

The leakage limits of LCO 3.6.1.2 will continue to be met prior to entering into OPERATIONAL CONDITIONS 1, 2, or 3 (i.e., temperature greater than 200 °F). Satisfying these leakage limits provides assurance that the measured leakage rate will not exceed the maximum allowable leakage rate of La during plant operation. Therefore, operation with the proposed change will 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: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

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

NRC Project Director: Susan Frant Shankman, Acting

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of amendment request: January 17, 1996.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) including revisions to Specifications 3/4.3.1, ''Reactor Protection System Instrumentation,'' 3/4.3.2, ''Isolation Actuation Instrumentation,'' 3/4.3.3, ''Emergency Core Cooling System Actuation Instrumentation,'' 3/4.3.4.2, ''End-of-Cycle Recirculation Pump Trip System Instrumentation,'' and the associated Bases to relocate response time limit tables from the TSs to the Updated Safety Analysis Report (USAR). The proposed revisions to the TSs also include several administrative 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:

1. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment relocates Tables 3.3.1–2, ''Reactor Protection System Response Times,'' 3.3.2–3, ''Isolation System Instrumentation Response Times'' 3.3.3–3, ''Emergency Core Cooling System Response Times'' and 3.3.4.2–3 ''End-of-Cycle Recirculation Pump Trip System Response Time'' from the Technical Specifications to the USAR. The Technical Specification Surveillance Requirements and associated actions are not affected and remain in the Technical Specifications. This change to the reactor protection system instrumentation, isolation actuation instrumentation, and emergency core cooling system instrumentation is being done in accordance with the guidance provided in Generic Letter 93–08, ''Relocation of Technical Specification Tables of Instrument Response Time Limits,'' and the change to the end-of-cycle recirculation pump trip system instrumentation is consistent with NUREG 1433, ''Standard Technical Specifications, BWR/4.'' This change allows NMP2 [Nine Mile Point Unit 2] to administratively control subsequent changes to the response time limits in accordance with 10CFR50.59. Additionally, procedures which contain the various response time limits are also subject to the change control provisions of 10 CFR 50.59. Relocating this information does not affect the initial conditions of a design basis accident or transient analysis. The proposed Technical Specification changes do not affect the capability of the associated systems to perform their intended functions within their required response times. Since any subsequent changes to the USAR or procedures which contain the response time limits are evaluated in accordance with 10CFR50.59, the proposed amendment does not involve an increase in the probability or consequences of an accident previously evaluated.

2. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed change would relocate the response time limit tables from the Technical Specifications to the USAR. Subsequent changes to the USAR, or in procedures which contain the various response time limits, would be evaluated in accordance with the requirements of 10CFR50.59, which would evaluate the possibility of the creation of a new or different kind of accident. The proposed change does not involve any physical alteration of the plant, change in a Limiting Condition for Operation or change in 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 previously evaluated.

3. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The proposed change would relocate the response time limit tables from the Technical Specifications to the USAR. Future changes to the response time limits in the USAR, or in procedures which contain the various response time limits, would be in accordance with 10CFR50.59, which would evaluate the proposed change to determine whether it involved any reduction in the margin of safety. The response time limits to be transposed from the Technical Specifications to the USAR are the same as the existing Technical Specifications. Therefore, this proposed 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 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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Project Director: Susan Frant Shankman, Acting.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of amendment request: January 25, 1996.

Description of amendment request: The proposed amendment would change a footnote in Table 3.3.3–1 and the corresponding footnote in surveillance Table 4.3.3.1–1 (both referenced by Technical Specification 3/4.3.3 ''Emergency Core Cooling System Actuation Instrumentation'') to more clearly define when, during cold shutdown and refueling (i.e., Operational Conditions 4 and 5), the Loss of Voltage and Degraded Voltage relays associated with the 4.16 kV Emergency Bus Undervoltage are required to be operable. The footnotes currently state: ''Required when ESF [Engineered Safety Features] equipment is required to be OPERABLE.'' The proposed amendment would change the footnotes to state: ''Required when the associated diesel generator is required to be OPERABLE.''

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 operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequence of an accident previously evaluated.

The proposed change would require the Loss of Power instruments to be OPERABLE in Operational Conditions 4 and 5 only when the associated diesel generator is required to be OPERABLE. The Loss of Power relays provide a support function to initiate the associated diesel generator start and bus unloading sequences. If that diesel generator is not in service, the loss of power relays perform no safety function. Therefore, relating diesel generator OPERABILITY and Loss of Power instrument OPERABILITY will not involve an increase in the probability of an accident previously evaluated.

The proposed change does not affect the requirements of ESF OPERABILITY. The change does not affect diesel generator response to a loss of voltage or degraded voltage on the Divisional 4.16 kV electrical busses when the diesel generator is required to be OPERABLE. Automatic response of the ESF functions is unaffected by removing the Loss of Power relays from service under these conditions, therefore, the proposed change will not involve a significant increase in the consequences of an accident previously evaluated.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve a modification of plant equipment nor does it change the way the equipment will be maintained or operated. The revision to Technical Specifications will continue to require the Loss of Power instrumentation to be OPERABLE when the associated diesel generator is required to be OPERABLE. The Loss of Power instruments will continue to perform their safety function of initiating the diesel generator start and bus unloading sequences.

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

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The proposed change will not affect the OPERABILITY, operation or reliability of any ESF function including the diesel generators. All ESF functions will remain available during postulated accidents with a loss of offsite electrical power. The change simply clarifies when the Loss of Power instruments are required to be OPERABLE during Operational Conditions 4 and 5. Therefore, the proposed change will 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: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

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

NRC Project Director: Susan Frant Shankman, Acting.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of amendment request: March 15, 1996.

Description of amendment request: The proposed amendment would revise the surveillance requirements of Technical Specification (TS) 4.6.2.1 ''Containment Systems—Depressurization Systems—Suppression Pool'' to extend the time interval for performing the containment drywell-to-suppression chamber bypass leakage test from 18 months to an interval corresponding to that required for the Containment Integrated Leak Rate Test. The provisions of TS 4.0.2 (which would provide an extension of up to 25% of the specified surveillance interval) will not apply. Specifically, existing TS 4.6.2.1.d would become subparagraphs d and e to require that the suppression pool be demonstrated operable:

d. At least once per 18 months by conducting a visual inspection of the exposed accessible interior and exterior surfaces of the suppression chamber.*

e. At least every outage by requiring the performance of a Containment Integrated Leak Rate Test, as scheduled in conformance with the criteria specified in the 10 CFR 50 Appendix J Testing Program Plan described in Section 6.8.4.f, by conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of 3 psi and verifying that the [drywell-to-suppression chamber bypass flow area] A/the square root of K calculated from the measured leakage is within the specified limit of 0.0054 square feet.

1. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission.

2. If two consecutive tests fail to meet the specified limit, a test shall be performed at least each refueling outage until two consecutive tests meet the specified limit, at which time the original test schedule may be resumed.

3. The provisions of Specification 4.0.2 do not apply.

*Includes each vacuum relief valve and associated piping.

The proposed changes would also add a new surveillance requirement for the testing of the bypass leakage path containing the suppression chamber vacuum breakers, with associated acceptance criteria, which would be performed each refueling outage that the bypass leak test is not performed. Specifically, a new TS 4.6.2.1f would require that the suppression pool be demonstrated operable:

f. During each refueling outage for which the drywell-to-suppression chamber bypass leak test in Specification 4.6.2.1.e is not conducted, by conducting a test of the four drywell-to-suppression chamber bypass leak paths containing the suppression chamber vacuum breakers at a differential pressure of at least 3 psi and

1. Verifying that the total leakage area A/the square root of K contributed by all four bypass leak paths is less than or equal to 24% of the specified limit, and

2. The leakage area for any one of the four bypass leak paths is less than or equal to 12% of the specified limit.

By separate action, the NRC has provided notice of a proposed amendment to change the frequency of Containment Integrated Leak Rate Tests in accordance with Option B of 10 CFR Part 50 Appendix J . The proposed changes described herein are intended to be consistent with the changes proposed under Option B.

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 operation on Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS changes involve the drywell-to-suppression chamber bypass leak test frequency. There are no physical or operational changes to the plant as a result of these proposed TS revisions. Furthermore, the primary containment acts as an accident mitigator and not as an accident initiator. Therefore, the proposed TS changes do not affect the probability of any previously evaluated accident.

The continued testing of bypass leakage pathways containing the suppression chamber vacuum breakers on a refueling frequency, and the continued requirement for visual inspection of containment structural features assures that the bypass leakage path will not degrade beyond the TS allowable limit during the interval between performance of the bypass leakage test. Therefore, radioactivity release following an accident will not be increased since the pressure suppression capability of the containment is not reduced from the existing design, and there will be no significant increase in the consequences of any accident previously evaluated.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed TS changes involve the drywell to suppression chamber bypass leak test frequency. There are no physical or operational changes as a result of these proposed TS changes. These proposed TS changes also include a requirement to continue performing a surveillance test on the bypass leakage pathways containing the vacuum breaker assemblies each refueling outage for which the drywell-to-suppression chamber test is not conducted. This test, along with the visual inspection required every refueling cycle, will ensure that acceptable bypass leakage is maintained during those intervals when the bypass leak test is not required. Accordingly, the possibility of a new or different type of accident is not introduced. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The drywell-to-suppression chamber bypass leak test data obtained during previous testing at NMP2 [Nine Mile Point Unit 2] demonstrates conformance, by a large margin, to the TS and design leakage requirements. The test data and engineering evaluations indicate that there is negligible risk that the bypass leakage will change adversely in future years. Furthermore, the proposed test frequency is judged to be acceptable based on the small risk of bypass leakage through paths other than those containing the suppression chamber vacuum breakers.

A test of the bypass leak pathways containing the vacuum breakers will be used to verify acceptable bypass leakage during those outages when the bypass leak test is not performed. The proposed test of the bypass leak pathways containing the vacuum breakers, with stringent acceptance criteria, combined with the other negligible potential leakage areas provide an acceptable level of assurance that the bypass leakage can be measured. This capability ensures that an adverse condition can be detected and corrected such that the existing level of confidence that the primary containment will function as required during a LOCA [loss-of-coolant accident] is maintained. Therefore, the proposed TS changes do not involve 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 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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Project Director: Susan Frant Shankman, Acting.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of amendment request: March 20, 1996.

Description of amendment request: The proposed amendment would revise Tables 3.3.1–1 and 4.3.1–1 of Technical Specification 3/4.3.1 ''Reactor Protection System Instrumentation'' to delete the operability requirement for the Average Power Range Monitor (APRM) Neutron Flux-Upscale, Setdown and Inoperative functions in Operational Conditions (OCs) 3 (Hot Shutdown) and 4 (Cold Shutdown). These same functions would also be revised for OC 5 (Refueling) to indicate that operability will only be required during shutdown margin demonstrations performed per TS 3.10.3.

Basis for proposed no significant hazards consideration determination: The revisions to the APRM functions are proposed to support licensee's plans to replace Local Power Range Monitors during the next refueling outage. The revisions also provide for the eventual replacement of the existing APRM System with the Nuclear Measurement Analysis and Control Power Range Neutron Monitoring System, and the eventual installation of the Oscillation Power Range Monitor system for the detection of reactor instability conditions. These modifications are based upon Report NEDO–31960, ''BWR Owners' Group Long-Term Solutions Licensing Methodology, approved by the Commission July 12, 1993; the licensee's response of November 8, 1994, selecting Option III in NEDO–31960 for Nine Mile Point, Unit 2; NRC Generic Letter 94–02, ''Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors'' dated July 11, 1994; and General Electric Licensing Topical Report, NEDC–32410P–A, ''Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC–PRNM) Retrofit Plus Option III Stability Trip Function,'' which was approved by the Commission September 5, 1995.

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 operation of Nine Mile Point Unit 2 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Reactor Protection System (RPS) initiates a reactor scram when one or more monitored parameters exceed their specified limits to preserve the integrity of the fuel cladding and the Reactor Coolant System and to minimize the energy that must be absorbed following a loss-of-coolant accident. The proposed changes will revise the OCs in which the APRM Neutron Flux-Upscale, Setdown and Inoperative RPS Instrumentation is required. These changes do not affect the probability of precursors of any accidents previously evaluated, and therefore, do not increase their probability.

During normal operation in OCs 3 and 4, all control rods are fully inserted and the reactor mode switch position control rod withdrawal blocks do not allow control rods to be withdrawn. Therefore, the RPS APRM functions are not required. Specification 3.9.10 does allow one control rod to be removed from the core in OC 4 by placing the mode switch in the refuel position. However, with the reactor mode switch in the refuel position, refueling interlocks are in place (i.e., one-rod out, etc.), which together with adequate shutdown margin will preclude unacceptable reactivity excursions. The APRM Neutron Flux-Upscale, Setdown function is not required during OC 5 except during shutdown margin demonstrations. The SRMs [source range monitors], IRMs [intermediate range monitors], and refueling interlocks provide adequate protection from reactivity excursions during OC 5. The exception is during the shutdown margin demonstration when more than one control rod will be withdrawn and the APRMs will continue to be required to be operable as a backup to the IRMs. Testing of the RPS APRM functions will continue to be performed in those OCs for which operability is required. Consequently, the reliability and performance of the RPS APRM functions in these OCs will not be adversely affected. Therefore, the proposed change will not result in a significant increase in the consequences of any accidents previously evaluated.

The operation of Nine Mile Point Unit 2 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes will revise the applicable OCs in which the APRM neutron Flux-Upscale, Setdown and Inoperative RPS instrumentation is required. Changes to OC requirements will not introduce any new accident precursors and will not involve any physical alternations to plant configurations which could initiate a new or different kind of accident. NMP2 is analyzed for a single control rod withdrawal error during refueling. Since the core is designed to meet shutdown requirements with the highest worth rod withdrawn, the core remains subcritical even with one rod withdrawn. The one-rod-out interlock which allows only one control rod to be withdrawn in OC 5 is not affected by the proposed changes. Consequently, the proposed changes do not create an accident different than the previously analyzed single control rod withdrawal error event. Surveillance testing will continue to be performed to assure reliability and maintain current performance levels. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

The operation of Nine Mile Point Unit 2 in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

The proposed changes to the RPS APRM function instrumentation Technical Specification requirements will not adversely affect the design or the performance characteristics of the RPS instrumentation nor will it affect the ability of the RPS APRM instrumentation to perform its intended function. As discussed above, the subject RPS instrumentation is not required in OC 3, 4, and 5 except for shutdown margin demonstrations. Accordingly, deletion of the requirement to have these functions operable in these OCs will not significantly reduce a margin of safety. Surveillance testing will continue to be performed for those OCs in which the instrumentation is required to assure 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 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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Project Director: Susan Frant Shankman, Acting.

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

Date of amendment request: March 28, 1996.

Description of amendment request: The proposed amendment would change Technical Specification Section 3.7.7, ''Sealed Source Contamination,'' by making the criteria for testing sealed sources for contamination and leakage at Millstone Unit No. 2 the same as those at Millstone Unit No. 3, the Haddam Neck Plant, and Seabrook Station. Specifically, the sealed sources that are required to be free of greater than or equal to 0.005 microcuries of removable contamination would be those that would exceed ''100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material.'' The Bases Section 3/4.7.7, ''Sealed Source Contamination,'' would also be changed to reference the appropriate section of 10 CFR 70.39.

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 (SHC), which is presented below:

Pursuant to 10 CFR 50.92, NNECO [Northeast Nuclear Energy Company] has reviewed the proposed changes and concludes that the changes do not involve a significant hazards consideration (SHC) since the proposed changes satisfy the criteria in 10 CFR 50.92(c). That is, the proposed changes do not:

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

The changes make the criteria for testing sealed sources for contamination and leakage at Millstone Unit No. 2 the same as those at Millstone Unit No. 3, the Haddam Neck Plant and Seabrook Station. Although the leakage criteria for sealed sources that are to be tested is being changed, the allowable leakage remains small. Any leakage that is identified would not cause a significant radiation exposure. The source storage area is routinely surveyed by Health Physics in accordance with Health Physics Department procedures and any significant leakage would be detected. Therefore, this change 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 from any accident previously evaluated.

The proposed change in the criteria for testing sealed sources for contamination and leakage will not change the way the sources are used. Therefore, this change will 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 possible radiation exposure to both the workers and the public from this change is very small. All protective systems which would detect any release of material from the site remain in place so there is no reduction in safety for the public. Likewise, all protective systems for the workers remain in place. Workers using the sources routinely pass through the whole body contamination monitors. In addition, the source storage areas are surveyed routinely by Health Physics in accordance with Health Physics Department procedures, and any significant leakage would be detected. The bases section is being revised to reference the appropriate section of 10 CFR 70.39. Therefore, there is no 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 Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360, and Waterford Library, Attn: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Project Director: Phillip F. McKee.

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: March 12, 1996.

Description of amendment request: The proposed changes would remove a requirement to interconnect two or more accumulators for the purpose of cross checking instrumentation in the event that one of the two pressure or level instrument channels on an accumulator is declared inoperable.

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

Response: The design basis accident for which the accumulators were designed is the double ended guillotine break of a cold leg. Interconnecting or not interconnecting accumulators does not have any effect on the probability of occurrence of this event. By eliminating the requirement to interconnect accumulators, the proposed amendment assures that a minimum of three accumulators are available, as assumed in the safety analyses, to mitigate the consequences of a large-break loss-of-coolant [LBLOCA] accident. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

Response: The proposed amendment does not involve any physical changes to plant equipment or setpoints and does not create the possibility of a new or different kind of accident. Eliminating the requirement to interconnect accumulators ensures that the plant configuration is maintained consistent with that assumed in the safety analysis and no new failure modes are created.

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

Response: There is no margin of safety specified in the Technical Specifications for these instrument channels. There are no setpoints or allowable values associated with these instrument channels which affect Safety Limits or Limiting Safety System Settings. The proposed amendment ensures that the safety analysis assumption regarding the accumulators remains valid and the resulting peak fuel clad temperature meets specified acceptance criteria. The proposed 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 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: Susan Frant Shankman, Acting.

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: March 14, 1996.

Description of amendment request: The proposed changes would allow a one-time extension of the inspection interval for the steam generator tubes that is due in July 1996.

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

Response: The proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. As stated in the Basis of the IP3 [Indian Point Unit 3] Technical Specifications, the program for inservice inspection of steam generator tubes regarding equipment, procedures, and sample selection is based upon the guidance and recommendations in Regulatory Guide 1.83 and NRC Generic Letter 85–02. The addition of the footnote to extend the surveillance due date will not increase the deviation from the guidance and recommendation stated above, and, therefore will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve the addition of any new or different type of equipment, nor does it involve the operation of equipment required for safe operation of the facility in a manner different from those addressed in the Final Safety Analysis Report. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: The proposed license amendment does not involve a significant reduction in a margin of safety. The proposed change does not adversely affect any safety related system or component operation or operability, instrument operation, or safety system setpoints and does not result in increased severity of any of the accidents considered in the safety analysis. This change has no adverse effect on any margin of safety and, therefore, does not create 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: 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: Susan Frant Shankman, Acting

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: March 22, 1996.

Description of amendment request: The amendment proposes changes to the Technical Specifications (TS) to establish operability requirements for avoidance and protection from thermal hydraulic instabilities to be consistent with Boiling Water Reactor Owners Group long-term solution Option I–D. Editorial changes are also made to support the revised specifications, improve readability of Bases sections, and enhance the presentation of requirements for single loop 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:

Operation of the FitzPatrick 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 because:

The implementation of BWR Owners' Group long-term stability solution Option I–D at FitzPatrick does not modify the assumptions contained in the existing accident analysis. The use of an exclusion region and the operator actions required to avoid and minimize operation inside the region do not increase the possibility of an accident. Conditions of operation outside of the exclusion region are within the analytical envelope of the existing safety analysis. The operator action requirement to exit the exclusion region upon entry minimizes the possibility of an oscillation occurring. The actions to drive control rods and/or to increase recirculation flow to exit the region are maneuvers within the envelope of normal plant evolutions. The flow referenced scram has been analyzed and will provide automatic fuel protection in the event of an instability. Thus, each proposed operating requirement provides defense in depth for protection from an instability event while maintaining the existing assumptions of the accident analysis.

2. Create the possibility of a new or different kind of accident from those previously evaluated because:

The proposed operating requirements either mandate operation within the envelope of existing plant operating conditions or force specific operating maneuvers within those carried out in normal operation. Since operation of the plant with all of the proposed requirements are within the existing operating basis, an unanalyzed accident will not be created through implementation of the proposed change.

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

Each of the proposed requirements for plant thermal hydraulic stability provides a means for fuel protection. The combination of avoiding possible unstable conditions and the automatic flow referenced reactor scram provides an in depth means for fuel protection. Therefore, the individual or combination of means to avoid and suppress an instability supplements 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 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: Susan Frank Shankman, Acting.

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: March 22, 1996.

Description of amendment request: The amendment proposes to revise Technical Specification (TS) Table 3.2–2, ''Core and Containment Cooling System Initiation and Control Instrumentation Operability Requirements.'' The proposed changes will revise allowed outage times (AOTs) for 4kV Emergency Bus Undervoltage Trip Functions. The AOTs for these trip functions were extended by Amendment 227; however, the AOT extensions for these trip functions were not consistent with the requirements of Standard Technical Specifications (STS), NUREG–1433, and differed from the recommendations in the associated Licensing Topical Report. Additional changes are proposed to TS Table 3.2–2 and to TS Table 4.2–2, ''Core and Containment Cooling System Instrumentation Test and Calibration Requirements.'' These changes will: (1) replace the generic actions for inoperable instrument channels with function-specific actions, (2) replace the generic test AOT with function-specific test AOTs, and (3) relocate selected trip functions from the TS to an Authority 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:

Operation of the FitzPatrick 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.

The proposed changes are limited to replacement of the generic actions and test AOT with function-specific actions and test AOTs, and relocation of selected trip functions from the TS to an Authority controlled document. The changes do not introduce any new modes of plant operation, make any physical changes, or alter any operational setpoints. Therefore, the changes do not degrade the performance of any safety system assumed to function in the accident analysis. Consequently, there is no effect on the probability or consequences of an accident.

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

The proposed changes do not introduce any new accident initiators or failure mechanisms since the changes do not introduce any new modes of plant operation, make any physical changes, or alter any operational setpoints. Therefore the changes do not create the possibility of a new or different kind of accident.

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

The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The relocated requirements do not satisfy the 10 CFR 50.36 criteria for inclusion in the Technical Specifications. Therefore, the changes do not involve 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 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: Susan Frant Shankman, Acting.

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: March 27, 1996.

Description of amendment request: The amendment proposes to revise the Technical Specifications to support adoption of the primary containment leakage rate testing requirements of Option B to 10 CFR 50, Appendix J at the FitzPatrick plant, and clarify the numerical value of the allowable containment leakage rate (L_a) as 1.5 percent per day.

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 Authority has evaluated the proposed TS Amendment and determined that it does not represent a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, operation of the James A. FitzPatrick Nuclear Power Plant in accordance with the proposed amendment will not:

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

The proposed changes do not involve a change to the design or operation of the plant. The systems affected by this proposed TS change are not assumed in any safety analyses to initiate any accident sequence. Therefore, the probability of any accident previously evaluated is not increased by this proposed TS change. The clarification of the allowable containment leakage rate (L_a) is consistent with the accident analyses. There is no change to the consequences of an accident previously evaluated because maintaining leakage within limits assumed in the accident analyses ensures that the dose consequences resulting from an accident are not increased. The proposed TS changes maintain an equivalent level of reliability and availability for all affected systems. The ability of the affected systems associated with maintaining leak rate integrity to perform their intended function is unaffected by the proposed TS changes. Implementation of these changes will provide continued assurance that specified parameters associated with containment integrity will remain within acceptance limits, and as such, will not significantly increase the consequences of a previously evaluated accident.

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

The proposed changes allow adoption of those requirements specified in Option B to 10 CFR 50, Appendix J, and do not involve a change to the plant design and operation. As a result, the proposed changes do not affect the parameters or conditions that could contribute to the initiation of any accidents. The methods of performing primary containment leakage rate testing are not changed. No new accident modes are created by allowing extended intervals for Type A, B and C testing, or by clarifying the numerical value of the allowable containment leakage rate (L_a). No safety-related equipment or safety functions are altered, or adversely affected, as a result of these changes. The proposed changes will not introduce failure mechanisms beyond those already considered in the current plant safety analyses. Extension of the test intervals, and clarification of the allowable leakage rate, does not contribute to the possibility of a new or different kind of accident or malfunction from those previously analyzed.

3. Involve a significant reduction in the margin of safety because: The proposed changes affect the frequency of primary containment leakage rate testing, and the numerical definition of the allowable containment leakage rate (L_a). The design of the FitzPatrick plant is not changed. The methodology for test performance is unchanged and Type A, B and C tests will continue to be performed at $\geq P_a$. The proposed changes provide sufficient controls to ensure that proper maintenance and repairs are performed on the primary containment, and systems and components penetrating the primary containment. The reliability of containment systems assumed to operate in the plant safety analyses is not reduced. The numerical value of L_a specified in Specification 6.20 is consistent with the accident analyses, therefore, the dose consequences of any analyzed accidents are not increased. Therefore, the proposed changes provide continued assurance of the leak tightness of the containment without adversely affecting the public health and safety and, as such, will not involve 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 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: Susan Frant Shankman, Acting.

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 22, 1996.

Description of amendment request: The amendments would change the Technical Specifications to implement 10 CFR Part 50, Appendix J, Option B, for the Type A test by referring to Regulatory Guide 1.163, ''Performance-Based Containment Leakage-Test Program.''

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.

Containment leak rate testing is not an initiator of any accident. The proposed changes do not make any physical changes to the containment. The proposed changes do not affect performance of the containment, reactor operations or accident analysis. Therefore, the proposed changes will not involve an increase in the probability of any previously evaluated accident.

Since the allowable leakage rate is not being changed and since the analysis documented in NUREG–1493, ''Performance-Based Containment Leak-Test Program'' concludes that the impact on public health and safety due to extended intervals is negligible, the proposed changes will not involve an increase in the consequences of any previously evaluated accident. Therefore, adoption of a performance-based verification of leakage rates for the overall containment boundary will provide an equivalent level of safety and does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change makes no physical changes to the plant. Since no physical changes are involved and since the analysis documented in NUREG–1493 confirms that the performance based schedule continues to maintain a minimal impact on public risk, it can be concluded that the effect of the containment on any accident will not change. The proposed change does not affect normal plant operations or configuration, nor does it affect leak rate test pressure.

Therefore, the proposed amendment will 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.

The proposed changes are based on NRC-accepted provisions, and maintain necessary levels of reliability of containment integrity. The performance-based approach to leakage rate testing recognizes that historically good results of containment testing provide appropriate assurance of future containment integrity. This supports the conclusion that the impact on the health and safety of the public as a result of extended test intervals is negligible. Since the analysis documented in NUREG–1493 confirms that the performance based schedule continues to maintain a minimal impact on public risk, it can be concluded that the margin of safety is not significantly affected by the proposed changes.

The test history at Salem Units 1 and 2 (no ILRT failures) provides continued assurance of the leak tightness of the containment structure.

Therefore, the proposed amendment will 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: 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: John F. Stolz.

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

Date of amendment request: April 4, 1996 (TS 96–01).

Description of amendment request: The proposed change would revise the appropriate technical specifications, surveillances, and bases as needed for the conversion from Westinghouse nuclear fuel to Framatome Cogema Mark-BW17 nuclear fuel.

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:

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

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

The analyses provided in Topical Report BAW–10220P show that the changes do not significantly change the results of previously evaluated events. These analyses provide the template for accident analyses assumptions that must be met by the cycle-specific reload analysis.

The SQN Units 1 and 2 Cycle 9 reload cores with Mark-BW fuel will be designed to operate within the approved limits for accident analysis. The limits provided in the TS and described in the Updated Final Safety Analysis Report (UFSAR) provide the framework for accident analyses. By maintaining these limits, the probability or consequences of accidents related to the core changes do not significantly change. Thus, it is concluded that there is no significant increase in the probability or consequences of an accident previously evaluated.

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

The change to Mark-BW fuel cores and mixed (transition) cores has been evaluated in the Topical Report BAW–10220P. It was concluded that the change did not create new or different kinds of accidents. The change in fuel suppliers has been evaluated for consideration of the effects of power distribution and peaking factors such that there are no restrictions on the use of Mark-BW fuel assemblies beyond those already established in the UFSAR and TS. Adherence to the safety analysis limits restricts the possibility of new or different accidents. Historically, new accidents have not been associated with changes in fuel suppliers as long as safety analysis limits continue to be met. It is concluded that transition to Mark-BW fuel does not create the possibility of a new or different kind of accident from any previously analyzed.

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

The margin of safety is established by the acceptance criteria used by NRC. Meeting the acceptance criteria assures that the consequences of accidents are within known and acceptable limits. The loss-of-coolant accident (LOCA) acceptance criteria are unchanged: peak cladding temperature of ≤2200 degrees Fahrenheit, peak cladding oxidation of ≤17 percent, average clad oxidation of ≤1 percent, and long-term coolability. These requirements continue to be met. The methods used to demonstrate conformance with these limits have changed, and were reviewed to assure that the methods, as well as the results, are acceptable. The acceptance criteria for Departure from Nucleate Boiling (DNB) events has not changed and is still the 95 percent probability and 95 percent confidence interval that DNB is not occurring during the transient. The DNB correlation, and methods used to demonstrate that DNB limits are met, have changed, and these changes were reviewed to assure conformance with acceptable practices. Other changes, as well as the changes discussed above, have been evaluated in the referenced safety analyses and are shown to meet applicable acceptance criteria. Other margins, such as avoiding fuel centerline melting are not significantly changed. Based on these results, it is concluded that the margin of safety is not significantly reduced.

The NRC has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CPR 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: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

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

NRC Project Director: Frederick J. Hebdon.

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

Date of application request: April 12, 1996.

Description of amendment request: The proposed amendment would change Technical Specification (TS) 3/4.4 and its associated Bases to address the installation of laser welded tube sleeves in the Callaway Plant steam generators.

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 elevated tubesheet LWS [laser welded sleeve] configuration has been designed and analyzed in accordance with the requirements of the ASME [American Society of Mechanical Engineers] Code. The applied stresses and fatigue usage for the sleeve and weld are bounded by the limits established in the ASME Code. ASME Code minimum material property values are used for the structural and plugging limit analysis. Ultrasonic inspection is used to verify that minimum weld fusion zone thickness are produced. Mechanical testing has shown that the individual joint structural strength of Alloy 690 LWS under normal, upset and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (3 times normal operating pressure differential) burst margin recommended by RG [Regulatory Guide] 1.121. Therefore, each individual joint provides for structural integrity exceeding RG recommendations.

Leakage testing for ⅞″ and ¾″ tube sleeves has demonstrated that no unacceptable levels of primary to secondary leakage are expected during any plant condition, including the case where the seal weld is not produced in the lower joint of the tubesheet sleeve. Similar tests of 11/16'' tube sleeves will be completed prior to Refuel 8.

The sleeve minimum acceptable wall thickness (used for developing the depth-based plugging limit for the sleeve) is determined using the guidance of Regulatory Guide 1.121 and the pressure stress equation of Section III of the ASME Code. The limiting requirement of Regulatory Guide 1.121, which applies to part throughwall degradation, is that the minimum acceptable wall must maintain a factor of safety of three against tube failure under normal operating (design) conditions. A bounding set of design and transient loading input conditions was used for the minimum wall thickness evaluation in the generic evaluation. Evaluation of the minimum acceptable wall thickness for normal, upset and postulated accident condition loading per the ASME Code indicates these conditions are bounded by the design condition requirement minimum wall thickness.

A bounding tube wall degradation growth rate per cycle and an eddy current uncertainty has been assumed for determining the sleeve TS plugging limit. The sleeve wall degradation extent determined by eddy current examination, which would require plugging sleeved tubes, is developed using the guidance of RG 1.121 and is defined in WCAP–14596 to be 39 percent throughwall of the sleeve nominal wall thickness.

The consequences of failure of the sleeve joint are bounded by the current steam generator tube rupture analysis included in the Callaway FSAR. Due to the slight reduction in diameter caused by the sleeve wall thickness, primary coolant release rates would be slightly less than assumed for the steam generator tube rupture analysis (depending on the break location), and therefore, would result in lower total primary fluid mass release to the secondary system.

The proposed change does not adversely impact any other previously evaluated design basis accident of the results of LOCA and non-LOCA accident analyses for the current TS minimum reactor coolant system flow rate. The results of the analyses and testing demonstrate that the sleeve assembly is an acceptable means of maintaining tube integrity. Furthermore, per Regulatory Guide 1.83, ''Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes'' recommendations, the sleeved tube can be monitored through periodic inspections with present eddy current techniques. These measures demonstrate that installation of sleeves spanning degraded areas of the tube will restore the tube to a condition consistent with its original design basis.

Corrosion testing of laser welded sleeve joints indicates that the corrosion resistance (relative to roll transition control samples) can be increased by greater than a factor of ten with the application of a post weld heat treatment [PWHT]. All free span laser welds will receive a post weld heat treatment. Therefore, rapid corrosion degradation of the free span laser weld joint region is not expected. Recently performed corrosion testing of LWS joints in locked (at the first TSP [tube support plate] structure) tube conditions indicates that the PWHT, the stress corrosion cracking initiation potential in the weld region of the parent tube is reduced and the cracking resistance is enhanced. Similar test results and conclusions would be expected for Callaway based on the similarity of designs and expected tube far field residual stresses.

Conformance of the sleeve design with the applicable sections of the ASME Code and results of the leakage and mechanical tests, support the conclusion that installation of LWS will not increase the probability or consequences of an accident previously evaluated.

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

Sleeving will not adversely affect any plant component. Stress and fatigue analysis of the repair has shown that the ASME Code and Regulatory Guide 1.121 criteria are not exceeded. Implementation of LWS maintains overall tube bundle structural and leakage integrity at a level consistent to that of the originally supplied tubing during all plant conditions. Leak and mechanical testing of sleeves support the conclusions of the calculations that each sleeve joint retains both structural and leakage integrity during all conditions. Sleeving of tubes does not provide a mechanism resulting in an accident outside of the area affected by the sleeves. Any accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing tube rupture accident analysis.

Implementation of LWS will reduce the potential for primary to secondary leakage during a postulated steam line break while not significantly impacting available primary coolant flow area in the event of a LOCA. By effectively isolating degraded areas of the tube through repair, the potential for steam line break leakage is reduced. These degraded intersections now are returned to a condition consistent with the Design Basis. While the installation of a sleeve reduces primary coolant flow, the reduction is far below that caused by plugging. Therefore, far greater primary coolant flow area is maintained through sleeving versus plugging.

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

The LWS repair of degraded steam generator tubes has been shown by analysis to restore the integrity of the tube bundle consistent with its original design basis condition, i.e., tube/sleeve operational and faulted condition stresses are bounded by the ASME Code requirements and the repaired tubes are leaktight. The safety factors used in the design of sleeves for the repair of degraded tubes are consistent with the safety factors in the ASME Code used in steam generator design. The design of the tubesheet sleeve lower joints for the 3⁄4'' and 7⁄8'' sleeves have been verified by testing to preclude leakage during normal and postulated accident conditions. Similar tests of 11⁄16'' sleeves will be completed prior to Refuel 8. The portions of the installed sleeve assembly which represent the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation, thus satisfying the requirements of Regulatory Guide 1.83. The portion of the tube bridged by the sleeve joints is effectively removed from the pressure boundary, and the sleeve then forms the new pressure boundary. The areas of the sleeved tube assembly which require inspection are defined in WCAP–14596.

In addition, since the installed sleeve represents a portion of the pressure boundary, a baseline inspection of these areas is required prior to operation with sleeves installed. The effect of sleeving on the design transients and accident analyses has been reviewed based on the installation of sleeves up to the level of steam generator tube plugging coincident with the minimum reactor flow rate and the Callaway Safety Analysis.

Provisional requirements cited in other NRC Safety Evaluation Reports addressing the implementation of sleeving have required the reduction of the individual steam generator normal operation primary to secondary leakage limit from 500 to 150 gpd. Consistent with these evaluations, Union Electric will reduce the per steam generator leak rate limit of 500 gpd in TS 3.4.6.2.c to 150 gpd. The establishment of this leakage limit at 150 gpd provides additional safety margin.

Finally, Union Electric will reduce the tube plugging limit from 48 percent through wall to 40 percent through wall to be consistent with NUREG–1431. The establishment of the plugging limit at 40 percent through wall provides additional 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: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Project Director: William H. Bateman.

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

Date of application request: April 12, 1996.

Description of amendment request: The proposed amendment would change Technical Specification (TS) 3/4.4 and its associated Bases to address the installation of electrosleeves in the Callaway Plant steam generators.

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 electrosleeve configuration has been designed and analyzed in accordance with the requirements of the ASME [American Society of Mechanical Engineers] Code. The applied stresses and fatigue usage for the sleeve are bounded by the limits established in the ASME Code. ASME Code minimum material property values are used for the structural and plugging limit analysis. Mechanical testing has shown that the structural strength of nickel electrosleeves under normal, upset and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (3 times normal operating pressure differential) burst margin recommended by RG [Regulatory Guide] 1.121. Leakage testing for 5⁄8″, 7⁄8″ and 3⁄4″ tube sleeves has demonstrated that no unacceptable levels of primary to secondary leakage are expected during any plant condition. Similar tests of 11⁄16″ tube electrosleeves will be completed prior to Refuel 8.

The sleeve nominal wall thickness (used for developing the depth-based plugging limit for the sleeve) is determined using the guidance of Regulatory Guide 1.121 and the pressure stress equation of Section III of the ASME Code. The limiting requirement of Regulatory Guide 1.121, which applies to part throughwall degradation, is that the minimum acceptable wall must maintain a factor of safety of three against tube failure under normal operating (design) conditions. A bounding set of design and transient loading input conditions was used for the minimum wall thickness evaluation in the generic evaluation. Evaluation of the minimum acceptable wall thickness for normal, upset and postulated accident condition loading per the ASME Code indicates these conditions are bounded by the design condition requirement minimum wall thickness.

A bounding tube wall degradation growth rate per cycle and an NDE [nondestructive examination] uncertainty has been assumed for determining the sleeve TS plugging limit. The sleeve wall degradation extent determined by NDE, which would require plugging sleeved tubes, is developed using the guidance of RG 1.121 and is defined in BAW–10219P to be 20 percent throughwall.

The consequences of failure of the sleeve are bounded by the current steam generator tube rupture analysis included in the Callaway FSAR [final safety analysis report]. Due to the slight reduction in diameter caused by the sleeve wall thickness, primary coolant release rates would be slightly less than assumed for the steam generator tube rupture analysis (depending on the break location), and therefore, would result in lower total primary fluid mass release to the secondary system.

The proposed change does not adversely impact any other previously evaluated design basis accident or the results of LOCA [loss-of-coolant accident] and non-LOCA accident analyses for the current TS minimum reactor coolant system flow rate. The results of the analyses and testing demonstrate that the electrosleeve is an acceptable means of maintaining tube integrity. Furthermore, per Regulatory Guide 1.83 recommendations, the sleeved tube can be monitored through periodic inspections with present NDE techniques. These measures demonstrate that installation of sleeves spanning degraded areas of the tube will restore the tube to a condition consistent with its original design basis.

Conformance of the electrosleeve design with the applicable sections of the ASME Code and results of the leakage and mechanical tests, support the conclusion that installation of electrosleeves will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Electrosleeving does not represent a potential to adversely affect any plant component. Stress and fatigue analysis of the repair has shown that the ASME Code and Regulatory Guide 1.121 criteria are not exceeded. Implementation of electrosleeving maintains overall tube bundle structural and leakage integrity at a level consistent to that of the originally supplied tubing during all plant conditions. Leak and mechanical testing of electrosleeves support the conclusions of the calculations that each sleeve retains both structural and leakage integrity during all conditions. Sleeving of tubes does not provide a mechanism resulting in an accident outside of the area affected by the sleeves. Any accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing tube rupture accident analysis.

Implementation of sleeving will reduce the potential for primary to secondary leakage during a postulated steam line break while not significantly impacting available primary coolant flow area in the event of a LOCA. By effectively isolating degraded areas of the tube through repair, the potential for steam line break leakage is reduced. These degraded intersections now are returned to a condition consistent with the Design Basis. While the installation of a sleeve reduces primary coolant flow, the reduction is far below that caused by plugging. Therefore, far greater primary coolant flow area is maintained through sleeving versus plugging.

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

The electrosleeve repair of degraded steam generator tubes has been shown by analysis to restore the integrity of the tube bundle consistent with its original design basis condition, i.e., tube/sleeve operational and faulted condition stresses are bounded by the ASME Code requirements and the repaired tubes are leaktight. The safety factors used in the design of sleeves for the repair of degraded tubes are consistent with the safety factors in the ASME Code used in steam generator design. The portions of the installed sleeve assembly which represent the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation, thus satisfying the requirements of Regulatory Guide 1.83. The portion of the tube bridged by the sleeve is effectively removed from the pressure boundary, and the sleeve then forms the new pressure boundary. The areas of the sleeved tube assembly which require inspection are defined in BAW–10219P.

In addition, since the installed sleeve represents a portion of the pressure boundary, a baseline inspection of these areas is required prior to operation with sleeves installed. The effect of sleeving on the design transients and accident analyses has been reviewed based on the installation of sleeves up to the level of steam generator tube plugging coincident with the minimum reactor flow rate and the Callaway Safety Analysis.

Provisional requirements cited in other NRC Safety Evaluation Reports addressing the implementation of sleeving have required the reduction of the individual steam generator normal operation primary to secondary leakage limit from 500 to 150 gpd.

Consistent with these evaluations, Union Electric will reduce the per steam generator leak rate limit of 500 gpd in TS 3.4.6.2.c to 150 gpd. The establishment of this leakage limit at 150 gpd provides additional safety margin.

Finally, Union Electric will reduce the tube plugging limit from 48 percent through wall to 40 percent through wall to be consistent with NUREG–1431. The establishment of the plugging limit at 40 percent through wall provides additional 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: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Project Director: William H. Bateman.

Vermont Yankee Nuclear Power Corporation, Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of amendment request: April 4, 1996.

Description of amendment request: The proposed amendment would revise the Technical Specifications regarding secondary containment integrity including addition of required actions in the event secondary containment integrity is not maintained when required. It would also require surveillance of the secondary containment isolation valves under the licensee's in-service testing program.

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 result in any hardware changes. The requirements for Secondary Containment integrity are not assumed in the initiation of any analyzed event. The proposed changes establish and maintain adequate assurance that Secondary Containment Integrity will be maintained as assumed in analyses for the mitigation of accident consequences. Not requiring Secondary Containment Integrity when the reactor coolant system is not vented in the Cold Shutdown condition or the Refuel Mode does not involve an increase in previously evaluated accident consequences since no mechanism exists to impart additional fission-products into the reactor coolant. Under these conditions, activities for which the reactor coolant system would not be vented would be strictly controlled and monitored. As a result, leaks or pipe breaks would typically be detected before significant inventory loss occurred. These activities would typically be performed after refueling when few noncondensible gases remain in the reactor coolant. The temperature limitation of 212°F will ensure that water, not steam, would be emitted from the postulated leak or pipe break. In addition, under these conditions, stored energy is sufficiently low that even with loss of inventory following a recirculation line break, core coverage would be maintained by the low pressure emergency core cooling systems required per Specification 3.5.H and the fuel would not exceed its peak clad temperature limit. As a result, the potential for failed fuel and a subsequent increase in reactor coolant activity is minimized and significant releases of radioactive material to the environment would not be expected to occur. Therefore, these changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal operation and will not alter the method used by any system to perform its design function. The proposed changes to not allow plant operation in any mode that is not already evaluated and will still ensure Secondary Containment Integrity is maintained when required. Thus, these changes do not create the possibility of a new or different kind of

accident from any accident previously evaluated.

(3) The proposed changes to Secondary Containment Integrity requirements have no impact on any safety analysis assumptions. Secondary Containment Integrity will be maintained as assumed in the safety analyses and as stated in current Bases 3.7.B and 3.7.C. Not requiring Secondary Containment Integrity when the reactor coolant system is not vented in the Cold Shutdown condition or the Refuel Mode does not involve significant reduction in a margin of safety since no mechanism exists to impart additional fission products into the reactor coolant. Under these conditions, activities for which the reactor coolant system would not be vented would be strictly controlled and monitored. As a result, leaks or pipe breaks would typically be detected before significant inventory loss occurred. These activities would typically be performed after refueling, at low decay levels, and with reactor coolant temperature less than or equal to 212°F. In addition, under these conditions, stored energy in the reactor core is very low. The reactor pressure vessel would rapidly depressurize in the event of a large primary system leak and the low pressure emergency core cooling systems required per Specification 3.5.H under these conditions would be adequate to keep the core flooded. This would ensure that the fuel would not be uncovered and would not exceed the peak clad temperature limit.

As a result, the potential for failed fuel and a subsequent increase in reactor coolant activity is minimized and significant releases of radioactive material to the environment would not be expected to occur. Therefore, these 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 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: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

Attorney for licensee: R.K. Gad, III, Ropes and Gray, One International Place, Boston, MA 02110–2624.

NRC Project Director: Susan Frant Shankman.

Vermont Yankee Nuclear Power Corporation, Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of amendment request: April 4, 1996.

Description of amendment request: The proposed amendment would revise the surveillance requirements for control rod over-travel to remove the specific testing methodology from the Technical Specifications to administratively controlled documents.

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 control rod drive mechanism over-travel is not considered to be the initiator of any previously analyzed accident. Verification of coupling of the control rods and drive mechanisms is performed by other means and continues to be required in the same manner, so there is no significant increase in the probability of a rod drop accident. The over-travel indication is also not considered in the mitigation of consequences of any previously analyzed accident, and the removal of a specific surveillance of the indication will not affect the response of the control rods or the reactor protection system to these accidents. Therefore, this change will not significantly increase the probability or consequences of any previously analyzed accident.

(2) The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) nor changes in parameters governing normal plant operation. The proposed change will continue to provide effective methods to assure the control rods and their drive mechanisms are coupled and preserve the safety functions associated with the prevention or automatic mitigation of design basis accidents. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) The proposed changes continue to provide an appropriate method for verification of the capability of the over-travel indication to perform its function. Therefore, this change will not significantly reduce 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: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

Attorney for licensee: R.K. Gad, III, Ropes and Gray, One International Place, Boston, MA 02110–2624.

NRC Project Director: Susan Frant Shankman.

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 15, 1996.

Description of amendment request: The proposed changes will clarify the applicability of the quadrant power tilt ratio (QPTR) 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:

Operation of Surry Power Station in accordance with the proposed Technical Specifications change will not:

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

The application of the QPTR limits, as proposed, will assure that the gross core radial power distribution remains consistent with design limits above 50% power. At or below 50% rated thermal power, there is insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require implementation of a QPTR limit on the distribution of core power. Therefore, the proposed change to clarify the applicability of the QPTR requirements has no impact on the probability of an accident occurrence and does not increase the consequences of any design basis accident.

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

There are no plant modifications or changes in methods of plant operation introduced by the proposed change. The change would limit the application of QPTR limits to operation at power levels >50% to preclude core power distributions from occurring which would violate fuel design criteria previously analyzed. At or below 50% rated thermal power, there is no impact to core power distributions which could affect the fuel design criteria. Therefore, the proposed change does not create the possibility for an accident or malfunction of a different type than that previously evaluated in the safety analysis report.

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

The proposed change only affects the applicability of the QPTR limits. The QPTR limits remain unchanged to preclude any violation of previously analyzed fuel design criteria. Adherence to the QPTR limits, hot channel factors, and applicable Limiting Conditions for Operation will continue. Therefore, the margin of safety as described in the Bases Section of any part of the Technical Specifications 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: 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: Eugene V. Imbro.

Previously Published Notices of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant 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.

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: April 3, 1996.

Description of amendment request: The proposed amendments would revise the hydrogen mitigation system Technical Specifications (TS). The change would provide that, if neither the Train A or Train B igniter is operable in any one containment region, then there is an allowance of 7 days to restore one hydrogen igniter to OPERABLE status, or be in Hot Shutdown within the next 6 hours. This would be consistent with the guidance of the Standard TS for Westinghouse plants, NUREG–0431.

Date of publication of individual notice in Federal Register: April 16, 1996 (61 FR 16649).

Expiration date of individual notice: May 16, 1996.

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

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

Date of amendment request: January 22, 1996, as supplement by letter dated April 4, 1996.

Brief description of amendments: The proposed amendment would modify the steam generator tube plugging criteria in Technical Specification 3/4.4.5, Steam Generators, and the allowable leakage in Technical Specification 3/4.4.6.2, Operational Leakage, and the associated Bases. The proposed amendment would allow the implementation of steam generator voltage-based repair criteria for the tube support plate (TSP)/tube intersections for Unit 1.

Date of individual notice in the Federal Register: April 16, 1996 (61 FR 16651)

Expiration date of individual notice: May 16, 1996.

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

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: March 14, 1996.

Description of amendment request: The proposed amendment would revise the Technical Specifications for Indian Point Nuclear Generating Unit No. 3 to allow a one-time extension of the test intervals for the pressurizer safety valve setpoint and snubber functional testing that is due in May 1996.

Date of publication of individual notice in Federal Register: April 3, 1996 (61 FR 14835)

Expiration date of individual notice: May 3, 1996.

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

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, and at the local public document rooms for the particular facilities involved.

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

Date of application for amendments: November 22, 1995.

Brief description of amendments: The amendments delete from the Technical Specifications certain review responsibilities of the Plant Operations Review Committee and the Site Operations Review Committee relating to the Emergency Plan and the Security Plan and their respective implementing procedures. The proposed changes are consistent with the guidance of Generic Letter 93–07.

Date of issuance: April 24, 1996

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

Amendment Nos.: 189, 94, 196, and 128

Facility Operating License Nos. DPR–61, DPR–21, DPR–65, AND NPF–49: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: February 14, 1996 (61 FR 5812)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 24, 1996.

No significant hazards consideration comments received: No.

Local Public Document Room location: Russell Library, 123 Broad Street Middletown, Connecticut 06457, for the Haddam Neck Plant, and the Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385, for Millstone 1, 2, and 3.

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

Date of application for amendments: January 11, 1996, as supplemented by letter dated April 2, 1996.

Brief description of amendments: The amendments revise Technical Specification Table 3.6–1, Table 3.6–2a and Table 3.6–2b to delete references to process penetration M308 and service water system (RN) valves RN–429A and RN–432B from the lists of secondary containment bypass valves and containment isolation valves. The RN valves are no longer in service and are planned to be removed in forthcoming outages. The penetration will then be capped with blank flanges.

Date of issuance: April 23, 1996

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

Amendment Nos.: 143 and 137

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

Date of initial notice in Federal Register: February 14, 1996 (61 FR 5813) The April 2, 1996, letter provided additional information that did not change the scope of the January 11, 1996, application and the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 23, 1996.

No significant hazards consideration comments received: No

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

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

Date of application of amendments: December 7, 1995.

Brief description of amendments: The amendments revise Secondary Decay Heat Removal Technical Specification (TS) 3.4.2 and TS Table 4.1–1 to delete the requirement of having the main feedwater pump discharge header pressure switch provide an input to actuate the Anticipatory Reactor Trip System and Emergency Feedwater System.

Date of Issuance: April 15, 1996.

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

Amendment Nos.: 216, 216, 213.

Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: January 22, 1996 (61 FR 1628).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 15, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: October 25, 1993, as supplemented August 31, 1994, and October 5, 1995.

Brief description of amendments: The amendments modify the surveillance requirements related to dune survey and mangrove swamp monitoring and relocate them to the Final Safety Analysis Report

Date of Issuance: April 11, 1996.

Effective Date: April 11, 1996.

Amendment Nos.: 142 and 82.

Facility Operating License Nos. DPR–67 and NPF–16: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: December 22, 1993 (58 FR 67844) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 11, 1996

No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003

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

Date of application for amendment: August 31, 1995, as supplemented February 29, 1996.

Brief description of amendment: The amendment revises License Condition 2.B(6)(c), Fire Protection, and relocates fire protection requirements from the Maine Yankee Atomic Power Station Technical Specifications to the Maine Yankee Fire Protection Plan. The amendment is consistent with the guidance of NRC Generic Letters 86–10, Implementation of Fire Protection Requirements, and 88–12, Removal of Fire Protection Requirements, from the Technical Specifications.

Date of issuance: April 5, 1996.

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

Amendment No.: 156.

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

Date of initial notice in Federal Register: October 11, 1995 (60 FR 52932) The February 29, 1996, letter provided document dates that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 5, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 25, 1995.

Brief description of amendment: The amendment changes the Technical Specification regarding the average power range monitor (APRM) setpoints. These changes establish limiting conditions for operations and surveillance requirements for the APRM flow-biased scram and rod block setpoints. The amendment also incorporates several editorial changes and renumbered pages, removal of blank pages, revised Table of Contents, and modified Bases section for APRM setpoint requirements.

Date of issuance: April 15, 1996.

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

Amendment No.: 93.

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

Date of initial notice in Federal Register: December 20, 1995 (60 FR 65682).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 15, 1996.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360 and at the temporary local public document room located at the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385.

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

Date of amendment request: August 4, 1995, as supplemented by letter dated January 22, 1996.

Brief description of amendment: This amendment revised the Technical Specifications (TS) for the requirements for the containment radiation high signal (CRHS) and the safety injection and refueling water (SIRW) tank low signal (STLS) contained in TS 2.15, Tables 2–3 and 2–4. Table 3–2 of TS 3.1 will also be revised to include administrative changes to the CRHS surveillance methods to be consistent with the applicable surveillance functions. The Basis of TS 2.15 is being revised to clarify that the number of installed channels for CRHS is two. The term ''SOURCE CHECK'' is being deleted from the Definitions section.

Date of issuance: April 24, 1996.

Effective date: April 24, 1996.

Amendment No.: 173.

Facility Operating License No. DPR–40. The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 30, 1995 (60 FR 45182).

The January 22, 1996, supplemental letter provided additional clarifying information and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 24, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 1, 1995, as supplemented by letter dated April 16, 1996.

Brief description of amendments: The amendments change the concentration of calibration gas required to calibrate the Hydrogen and Oxygen Analyzers, and support the requirements of Limerick Generating Station Transient Response Implementation Plan (TRIP) T–102, ''Primary Containment Control Bases.''

Date of issuance: April 23, 1996.

Effective date: Both units, as of date of issuance, to be implemented within 45 days.

Amendment Nos.: 116 and 78.

Facility Operating License Nos. NPF–39 and NPF–85. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: April 26, 1995 (60 FR 20525) The April 16, 1996 letter requested a new effective date and did not change the initial proposed no significant hazards consideration determination nor the Federal Register notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 23, 1996.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: August 21, 1992; supplemented September 3, 1993, and March 28, 1996 (TS 92–07).

Brief description of amendments: The amendments revise the allowable value for the reactor coolant system loss of flow reactor trip setpoint from greater than or equal to 89.4 percent to greater than or equal to 89.6 percent.

Date of issuance: April 26, 1996.

Effective date: April 26, 1996.

Amendment Nos.: 221 and 212.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the technical specifications.

Date of initial notice in Federal Register: September 30, 1992 (57 FR 45090). The September 3, 1993 and March 28, 1996 supplemental letters provided clarifying information which did not change the proposed no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 26, 1996.

No significant hazards consideration comments received: None.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

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 application for amendment: February 5, 1996.

Brief description of amendment: This amendment clarifies TS 3/4.3.2.1, Table 3.3–3, Safety Features Actuation System Instrumentation, and revises Bases 3/4.3.1 and 3/4.3.2, Reactor Protection System and Safety System Instrumentation, to accurately reflect the design and actuation logic of the diesel generator load sequencer and the essential bus undervoltage relays.

Date of issuance: April 23, 1996.

Effective date: April 23, 1996.

Amendment No.: 211.

Facility Operating License No. NPF–3: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: March 13, 1996 (61 FR 10397).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 23, 1996.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, Ohio 43606

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

Date of application for amendments: November 20, 1995, as supplemented March 14, 1996.

Brief description of amendments: These amendments would permit the use of 10 CFR Part 50 Appendix J, Option B, performance-based containment leakage rate testing.

Date of issuance: April 18, 1996.

Effective date: April 18, 1996.

Amendment Nos. 208 and 208.

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

Date of initial notice in Federal Register: December 20, 1995 (60 FR 65686) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 18, 1996.

No significant hazards consideration comments received: No.

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

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

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

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

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

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

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

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

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By June 7, 1996, 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 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. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–001, 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, 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 20555–001, 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 the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

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

Date of application for amendment: April 14, 1996.

Brief description of amendment: The proposed amendment clarifies operability requirements for reactor vessel water level instrumentation to permit testing of components required by technical specifications.

Date of issuance: April 16, 1996.

Effective date: April 16, 1996.

Amendment Nos.: 229, 244, and 204.

Facility Operating License Nos. DPR–33, DPR–52 and DPR–68: Amendment revises the technical specifications.

The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration, are contained in a Safety Evaluation dated April 16, 1996. Public comments requested as to proposed no significant hazards consideration: No.

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 application for amendment: April 18, 1996.

Brief description of amendment: The amendment approves the use of the station black out diesel generator in lieu of the emergency diesel generator associated with decay heat removal loop 2 during the tenth refueling outage. This condition will continue as long as no work is performed in the switchyard or on the SBODG or the remaining emergency diesel generator and a shutdown risk contingency plan is developed to ensure challenges to spent fuel pool cooling are minimized. This condition is expected to last for no more than seven days.

Date of issuance: April 19, 1996.

Effective date: April 19, 1996.

Amendment No.: 210.

Facility Operating License No. NPF–3: This amendment approved a one-time change to the design basis as described in the Updated Safety Analysis Report.

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

The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated April 19, 1996.

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West 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: Gail H. Marcus.

Dated at Rockville, Maryland, this 1st of May 1996.

For the Nuclear Regulatory Commission.

Steven A. Varga,

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

[FR Doc. 96–11295 Filed 5–7–96; 8:45 am]

BILLING CODE 7590–01–P

Privacy Act of 1974, as Amended; Establishment of a New System of Records

AGENCY: Nuclear Regulatory Commission.

ACTION: Establishment of a new system of records.

SUMMARY: The Nuclear Regulatory Commission (NRC) is proposing to establish a new Privacy Act System of Records, NRC–41, ''Tort Claims and Personal Property Claims,'' to maintain records needed to evaluate, settle, refer, pay, and/or adjudicate claims filed by individuals against the NRC.

EFFECTIVE DATES: The new system of records will become effective without further notice on June 17, 1996, unless comments received on or before that date cause a contrary decision. If changes are made based on NRC's review of comments received, NRC will publish a new final notice.

ADDRESSES: Send comments to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Docketing and Services Branch. Hand deliver comments to 11555 Rockville Pike, Rockville, Maryland, between 7:45 am and 4:15 pm Federal workdays. Copies of comments received may be examined, or copied for a fee, at the NRC Public Document Room at 2120 L Street, NW., Lower Level, Washington, DC.

FOR FURTHER INFORMATION CONTACT: Jona L. Souder, Freedom of Information/ Local Public Document Room Branch, Division of Freedom of Information and