specific and/or generic regulatory staff positions on the GDPs.

Although backfits are expected to occur and are a part of the regulatory process, it is important for sound and effective regulation that backfits are conducted in a controlled process. The NRC staff has developed NMSS Policy and Procedures Letter 1–53 on GDP generic and plant-specific backfitting. Copies of this procedure can be obtained from the Commission Public Document Room (PDR), 2120 L Street, NW., Washington, DC and at the Local Public Document Rooms (LPDRs), under Docket No. 70–7001, at the Paducah Public Library, 555 Washington Street, Paducah, Kentucky 42003; and under Docket No. 70–7002, at the Portsmouth Public Library, 1220 Gallia Street, Portsmounth, Ohio 45662.

Appendix 1 to NMSS Policy and Procedures Letter 1–53 provides guidance to the NRC staff on the proper NRC mechanisms (e.g., rulemaking) to use in establishing or communicating legal requirements and NRC staff positions to certificatees. Appendix 4 contains guidance to the NRC staff for making backfit determinations. Once a backfit determination has been made, and the proposed backfit does not meet either of the 2 exception [1] given in 10 CFR 76.76(a)(4) (i) and (ii), the NRC staff is required by 10 CFR 76.76(a)(3) to perform a cost/benefit analysis to determine ‘‘that there is a substantial increase (emphasis added) in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that plant are justified in view of this increased protection.’’

Appendix 3 of NMSS Policy and Procedures Letter 1–53 contains guidance on application of the ‘‘Substantial Increase’’ Standard. This standard provides qualitative criteria for NRC staff to make a safety/safeguards ‘‘net benefits’’ determination of cost/ benefits for the proposed backfit where a quantitative approach is not feasible.

NMSS Policy and Procedures Letter 1–53 is the first backfit procedure developed for facilities other than nuclear power reactor facilities. In addition, the GDPs are existing facilities which have operated under the Department of Energy for a number of years. Recognizing that this procedure may be addressing new issues, the NRC will accept public comments which focus on specific technical contents of the procedure.

[1] These exceptions are backfits that are necessary in order to ensure (a) that the plants provide adequate protection to the health and safety of the public and are in accord with the common defense and security, or (b) to bring the plants into compliance with the certificates, rules or orders of the Commission, or into conformance with written commitments by the Corporation.

Opportunity for Comments

The GDP backfit implementing procedure will be used by the NRC staff as an interim procedure pending completion of public review and resolution of comments on this FR Notice. Comments will be accepted which focus on the specific appendices discussed above. Comments in other areas of the procedures will be considered if they are directly related to the backfit issue. Procedures such as NMSS Policy and Procedures Letters are used by NRC as guidance to the NRC staff on NRC’s internal management process.

Dated at Rockville, Maryland this 17th day of March 1997.

For the Nuclear Regulatory Commission.

John T. Greeves,

Acting Director, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 97–7641 Filed 3–25–97; 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 March 3, 1997, through March 14, 1997. The last biweekly notice was published on March 12, 1997.

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 Chief, 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 April 25, 1997, 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.

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

Date of amendments request: January 15, 1997

Description of amendments request: The proposed change would revise the values of the minimum and maximum suppression pool water volumes corresponding to the upper and lower limits of the suppression water levels specified in TS 3.6.2.1.a.1 such that the implementation of the administrative controls will no longer be necessary to ensure compliance with the Technical Specifications.

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

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

The proposed change revises the values of the minimum and maximum suppression pool water volume limits. The water inventory of the suppression chamber is not a precursor of an accident and, therefore, cannot increase the probability of an accident previously evaluated. The pressure suppression chamber water pool mitigates the consequences of loss-of-coolant accidents (LOCAs), transients, and other events by providing a heat sink for reactor primary system energy releases. The proposed minimum and maximum pool water volume values will be consistent with the current suppression pool water level limits. No changes to setpoints will be made as a result of the proposed change. The impact of the proposed change to the minimum and maximum suppression pool volume limits on the suppression pool temperatures and pressures following a design basis LOCA, an SRV [Safety Relief Valve] blowdown event, an Anticipated Transient Without Scram (ATWS) event, an Appendix R fire event, and a station blackout event has been evaluated and does not cause accident parameters to exceed acceptable values. In addition, the impact the proposed change has on the time to reach cold shutdown when using the alternate RHR [Residual Heat Removal] shutdown cooling function is negligible.

The potential impact the proposed change to the suppression pool water volume limits has on SRV line loads, SRV discharge line reflood height, wetwell pressurization, suppression pool swell loads, vent thrust loads, and condensation oscillation and chugging loads was also reviewed. The proposed change to the suppression pool water volume limits has no adverse impact on any of these parameters.

The capability of the suppression chamber water pool to perform its mitigative functions is not affected by the proposed change. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change revises the values of the minimum and maximum volume of the suppression chamber water pool. The proposed change will not alter any physical mechanism by which the suppression chamber water pool volume is maintained between the minimum and maximum values. The suppression pool water level will continue to be maintained between -27 and -31 inches. As a result of the proposed change there are no physical changes to suppression chamber components or instrumentation. No new mode of operation is introduced as a result of the proposed change. Analyses have been performed which conclude that the proposed change would not affect the operability of equipment designed to mitigate the consequences of an accident. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed license amendment[s do] not involve a significant reduction in a margin of safety.

The proposed change revises the values of the minimum and maximum suppression chamber water pool volumes. The pressure suppression chamber water pool mitigates the consequences of several postulated accidents and transients by providing a heat sink for the primary coolant system. These accidents and events are the postulated design basis LOCA, Safety Relief Valve blowdown, ATWS, Appendix R fire and station blackout events. The consequences of the proposed change in the suppression pool water volume limits have been evaluated for these events.

The results of the analyses for the postulated accidents and events indicate the temperature of the suppression pool water could increase slightly as a consequence of the decrease in the minimum suppression pool water volume limit. However, the containment temperatures remain within acceptable values. The impact of the calculated increase in containment temperature on the available Net Positive Suction head (NPSH) for the Residual Heat Removal (RHR) and Core Spray pumps has been evaluated for the postulated design basis LOCA and indicate adequate NPSH is maintained throughout the event.

The potential impact of the proposed change to the suppression pool water volume limits on SRV line loads, SRV discharge line reflood height, wetwell pressurization, suppression pool swell loads, vent thrust loads, and condensation oscillation and chugging loads was evaluated with the conclusion that there are no adverse impacts on these parameters.

In addition, a small suppression pool water temperature increase could result due to the reduction in the minimum suppression pool volume limit in the event reactor shutdown is conducted through a path utilizing the suppression pool. Such a shutdown path is an alternative to the normal RHR shutdown cooling function, and the small potential increase in temperature results in a negligible increase in the time required to reach cold shutdown conditions. Cold shutdown conditions could still be reached well within the Technical Specification requirements.

The proposed increase in the suppression pool water volume limit does not adversely impact containment parameters as a result of postulated accidents and events. The potential increase in temperature of the pressure suppression pool water does not significantly decrease the ability to maintain containment parameters within acceptable limits. The potential increase in time to reach cold shutdown conditions utilizing the suppression pool as an alternative to the normal RHR shutdown cooling function is negligible. Therefore, the proposed change to revise the minimum and maximum suppression water pool volumes 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: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3297.

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: Mark Reinhart (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: March 14, 1997

Description of amendment request: The proposed change revises Technical Specification 3/4.5.4, ''Refueling Water Storage Tank,'' and its associated Bases.

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

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

The non-safety, non-seismic hydrotest pump is normally maintained separated from the RWST [Refueling Water Storage Tank] by a safety-related, locked closed manual operated boundary isolation valve (1CT-22). However, performance of Technical Specification required surveillance test OST-1506, ''Reactor Coolant System Isolation Valve Leak Test - 18 Month Interval- Mode 3,'' requires the short term use of the hydrotest pump during plant operating modes. Specifically, this hydrotest pump provides a high pressure source for leak testing the RCS [Reactor Coolant System] pressure isolation valves in Mode 3. The test is performed prior to entry into Mode 2, each refueling outage, whenever flow is established through the pressure isolation valves, or whenever the plant has been in cold shutdown for greater than 72 hours. Normally, the test is completed in less than 8 hours. Due to the piping configuration, a break in the non-seismic portion of the piping during these planned evolutions could result in draining the RWST below the minimum analyzed volume. Therefore to mitigate the consequences of a failure in the non-seismic piping, manual actions will be needed to isolate the break flow, (i.e., close valve 1CT-22), prior to reducing the water volume in the RWST below the minimum analyzed volume.

Based on the use of a dedicated attendant to close valve 1CT-22, the lack of significant accessibility concerns, and the reliability of the valve to function, it can be concluded that 30 minutes is ample time for a valve attendant stationed at the valve to execute the manual action. Since the RWST volume margin provides up to 103 minutes to respond to the pipe failure, it is reasonable to assume that manual actions to isolate the postulated pipe failure can be taken before the RWST level decreases below the minimum analyzed volume assumed in the safety analysis.

Therefore, there would be no increase in the probability or consequences of an accident previously evaluated.

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

Based on the use of a dedicated attendant to close valve 1CT-22, the lack of significant accessibility concerns, and the reliability of the valve to function, it can be concluded that 30 minutes is ample time for a valve attendant stationed at the valve to execute the manual action. Since the RWST volume margin provides up to 103 minutes to respond to the pipe failure, it is reasonable to assume that manual actions to isolate the postulated pipe failure can be taken before the RWST level decreases below the minimum analyzed volume assumed in the safety analysis. As a result, the capability of the RWST to perform its safety function is not impacted.

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

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

As described in the Technical Specification Bases, the operability of the RWST ensures that a sufficient supply of borated water is available for injection into the core by the emergency core cooling system. This borated water is used as cooling water for the core in the event of a LOCA [loss-of-coolant accident] and provides negative reactivty to counteract any positive increase in reactivity caused by reactor coolant system (RCS) cooldown. The limits on RWST minimum volume and boron concentration assure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all shutdown and control rods inserted except for the most reactive control assembly. These limits are consistent with the assumptions of the LOCA and steam line break analyses.

Based on the use of a dedicated attendant to close valve 1CT-22, the lack of significant accessibility concerns, and the reliability of the valve to function, it can be concluded that 30 minutes is ample time for a valve attendant stationed at the valve to execute the manual action. Since the RWST volume margin provides up to 103 minutes to respond to the pipe failure, it is reasonable to assume that manual actions to isolate the postulated pipe failure can be taken before the RWST level decreases below the minimum analyzed volume assumed in the safety analysis. As a result, the capability of the RWST to perform its safety function is not impacted.

Therefore, 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 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: Mark Reinhart, Acting

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

Date of amendment request: February 17, 1997

Description of amendment request: The proposed amendment would change the required diesel generator load during the initial 2 hours of a surveillance run from 2625 kW and 2750 kW to 2730 kW and 2860 kW.

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 of the following:

The proposed changes represent a correction to the emergency diesel generator surveillance requirement. The proposed changes are administrative in nature and do not significantly increase the probability or consequences of any previously evaluated accidents for Quad Cities Station. The proposed amendment is consistent with the current safety analyses and represents sufficient requirements for the assurance and reliability of equipment assumed to operate in the safety analysis. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

The associated systems related to this proposed amendment are not assumed in any safety analysis to initiate any accident sequence for Quad Cities Station; therefore, the probability of any accident previously evaluated is not increased by the proposed amendment.

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

The proposed amendment for Quad Cities Station's Technical Specification is required to ensure the diesel generator is tested in accordance with the design basis requirements. The proposed changes do not create the possibility of a new or different kind of accident previously evaluated for Quad Cities Station. No new modes of operation are introduced by the proposed changes. The proposed changes are administrative in nature and maintain at least the present level of operability. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The associated systems related to this proposed amendment are not assumed in any safety analysis to initiate any accident sequence for Quad Cities Station; therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed amendment is required to ensure the diesel generator is tested in accordance with the design basis requirements. The proposed changes are administrative in nature and do not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. The proposed changes have been evaluated and found to be acceptable for use at Quad Cities based on system design, safety analysis requirements and operational performance. Since the proposed changes are administrative in nature and maintain necessary levels of system or component reliability, the proposed changes do not involve a significant reduction in the margin of safety.

The proposed amendment for Quad Cities Station will not reduce the availability of systems required to mitigate accident conditions; therefore, the proposed 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 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: Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021

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

NRC Project Director: Robert A. Capra

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

Date of amendment request: December 24, 1996 and January 31, 1997

Description of amendment request: Changes to Administrative Controls section of the Technical Specifications needed to implement revised management responsibilities and titles that reflect the permanently shut down status of plant.

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

In accordance with 10 CFR 50.92, CYAPCO [Connecticut Yankee Atomic Power Company] and NNECO [Northeast Nuclear Energy Company] have reviewed the attached proposed changes and have concluded that they do not involve a Significant Hazard consideration (SHC). The basis of this conclusion is that the three criterion of 10 CFR 50.92 are not compromised. The proposed changes do not involve an SHC because the proposed changes will not:

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

No design basis accidents are affected by these proposed changes. The proposed changes are administrative in nature and are being proposed to reflect the organizational changes which became effective December 9, 1996.

The Haddam Neck unit changes are replacement of the Executive Vice President, Nuclear by the Executive Vice President and Chief Nuclear Officer along with the replacement of the Vice President, Haddam Neck by the Unit Director.

No safety systems are adversely affected by the proposed changes, and no failure modes are associated with the changes.

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

There are no changes in any way that the plants are operated due to this administrative change. The potential for an unanalyzed accident is not created. There is no impact on plant response, and no new failure modes are introduced. The proposed administrative and editorial changes have no impact on safety limits or design basis accidents, and have no potential to create a new or unanalyzed event.

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

These changes do not directly affect any protective boundaries nor do they impact the safety limits for the protective boundaries. These proposed changes are administrative and editorial in nature. Therefore there can be no reduction in the margin of safety.

The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (51 FR 7751, March 4, 1986) of amendments that are considered not likely to involve an SHC. The changes proposed herein are enveloped by example (1), since they are purely administrative changes to the technical specifications to reflect organizational title changes and to achieve consistence throughout the technical specifications at Haddam Neck.

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

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

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141-0270

NRC Project Director: Seymour H. Weiss

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

Date of amendment request: July 17, 1996

Description of amendment request: The proposed change request modifies Waterford Steam Electric Station, Unit 3, Technical Specifications 3/4.7.1.3,'' CONDENSATE STORAGE POOL,'' by increasing the minimum required contained water volume from 82 percent to 91 percent indicated level. This proposed change is required to ensure that the minimum useable water volume in the Condensate Storage Pool (CSP) is maintained greater than or equal to 170,000 gallons. The new minimum level accounts for the minimum level required to prevent Emergency Feedwater pump suction line vortexing and instrument measurement uncertainties.

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

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

Response: No.

Increasing the minimum required CSP level will insure that the minimum required 170,000 gallons of water is available for supply to the Emergency Feedwater System. Maintaining the minimum required water volume will not increase the probability of any accident previously evaluated. Additionally, it will not affect the consequences of any accident. Maintaining at least 170,000 gallons of water available in the CSP will ensure that the system remains within the bounds of the accident analysis. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different type of accident from any accident previously evaluated?

Response: No.

Increasing the minimum water volume of the CSP from 82 percent to 91 percent does not create a possibility for a new or different kind of accident. The CSP will be operated in the same manner as previously evaluated. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

Operation in accordance with this proposed change will ensure that the minimum contained water volume of the CSP will remain at least 170,000 gallons under all conditions. This will maintain the present margin of safety. 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 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street N.W., Washington, D.C. 20005-3502

NRC Project Director: William D. Beckner

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

Date of amendment request: February 5, 1997

Description of amendment request: The proposed amendment will change Waterford Steam Electric Station, Unit 3, Technical Specifications 3.1.2.7, 3.1.2.8, 3.5.1, 3.5.4, 3.9.1, and Bases 3/4.1.2. The proposed change will increase the minimum boron concentration in the Safety Injection Tanks (SITs) and the Refueling Water Storage Pool (RWSP) to 2050 ppm to reflect the safety analysis for fuel Cycle 9.

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 Safety Injection System (SIS) is designed to provide core cooling in the unlikely event of a loss of coolant accident (LOCA). The cooling must be sufficient to prevent significant alteration of core geometry, preclude fuel melting, limit the cladding metal-water reaction, and remove the energy generated in the core for an extended period of time following a LOCA. The SIS fluid must contain the necessary boron concentration to maintain the core subcritical for the duration of a LOCA.

The proposed change increases the minimum boron concentration in the SITs and RWSP from 1720 ppm to 2050 ppm. Thus, the SIT/RWSP will at all times contain sufficient borated water to provide adequate shutdown margin. Sampling of the system and RWSP required by the Technical Specifications assures that the required dissolved boron concentration is present. In addition to its emergency core cooling function, the SIS functions to inject borated water into the RCS to increase shutdown margin following a rapid cooldown of the RCS as a result of a steam line rupture.

Operation of the safety injection system is credited in the steam line break analysis for causing a decrease in core reactivity. The current minimum RWSP/SIT concentration to be injected is 1720 ppm. Thus an increase to 2050 ppm will have no adverse affect on this analysis.

The Mode 5 boron dilution event identifies that with an initial boron concentration of 1240 ppm, a Keff of 0.98, RCS partially drained, and one charging pump operational, the minimum possible time to criticality is greater than 90 minutes. For all other combinations of Keff, RCS conditions, and number of charging pumps, the time to loss of shutdown margin is greater than 55 minutes. Thus, the proposed increase in boron concentration will not affect the results of the Mode 5 boron dilution event.

The change to the action statement of TS 3.9.1 assures that the more limiting reactivity condition of a Keff less than 0.95 or a boron concentration of 2050 ppm specified in the COLR [Core Operating Limit Report] will be adhered to during refueling operations.

The upper limit on boron concentration has not changed; therefore, there will be no affect on boric acid precipitation post-LOCA.

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

The proposed change does not physically alter the configuration of the plant and, therefore, does not create the possibility of a new or different kind of accident from any previously evaluated accident.

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

The proposed change maintains the minimum of 55 minutes to criticality for the refueling mode boron dilution event analysis. The proposed change continues to ensure that borated water of sufficient concentration is injected from both the SITs and the RWSP in the event of a LOCA or MSLB [main steam line break] and that boric acid does not precipitate in the core during long term cooling following a LOCA.

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 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street N.W., Washington, D.C. 20005-3502

NRC Project Director: William D. Beckner

Nebraska Public Power District, Docket No. 50-298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: February 10, 1997

Description of amendment request: The proposed amendment would provide the requirements for avoidance and protection from thermal hydraulic instabilities as described in NRC Generic Letter 94-02, ''Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal Hydraulic Instabilities in Boiling Water Reactors.''

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

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

The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. In fact, it does not result in an increase in the probability or consequences of any previously evaluated accidents. The implementation of [Boiling Water Reactor Owners' Group] BWROG Long-Term Stability Solution Option I-D at [Cooper Nuclear Station] CNS 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-biased scram has been analyzed and will provide automatic fuel protection in the event of an instability. Thus, each proposed Technical Specification requirement provides defense for protection from an instability event within the existing assumptions of the accident analysis.

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

As stated above, the proposed Technical Specification 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 is within the existing operating basis, an unanalyzed accident will not be created through implementation of the proposed change.

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

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-biased 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 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: Auburn Memorial Library, 1810 Courthouse Avenue, Auburn, NE 68305

Attorney for licensee: Mr. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602-0499

NRC Project Director: William D. Beckner

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

Date of amendment request: March 6, 1997

Description of amendment request: During a self assessment, the licensee identified weaknesses in the current Technical Specifications regarding allowed outage times for certain specific protective instrumentation and also for reactor building access control. The proposed amendment is designed to eliminate these weaknesses by adopting guidance from NUREG-0123, ''Standard Technical Specifications for General Electric Boiling Water Reactors (BWR/

5),'' Revision 3, and NUREG-1433, Standard Technical Specifications General Electric Plants BWR/4,'' Revision 1.

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

1. The operation of Millstone Nuclear Power Station, Unit No. 1, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The inherent redundancy and reliability of the protective instrumentation trip systems ensure that the consequences of an accident are not significantly increased. In addition, the restrictive Allowable Outage Time (AOT) interval limits the probability of the protective instrument channel being unavailable and an accident requiring its function from occurring simultaneously. The requirement that the associated trip function maintains trip capability ensures that the protective instrumentation response will occur such that the consequences of an accident are not different from those previously evaluated.

Instruments addressed in the proposed TS respond to changes in the plant. The proposed (AOTs) provide a two-hour interval where the instrument is inoperable, yet the Technical Specification (TS) Limiting Condition for Operation (LCO) action statement is not immediately entered. The probability of a plant transient being initiated by a trip of a coincident channel during surveillance testing is reduced since the channel under test will only be tripped for a small portion of the test interval. Therefore, AOTs provided by the proposed TS have no effect on the probability of occurrence of previously evaluated accidents.

The proposed TS changes provide a two-hour interval where the instrument is inoperable, but the TS LCO action statement is not immediately entered. If a single failure occurred on the other channel of the trip system being tested and the channel being tested was not in the trip condition, a valid signal might not provide the required protective action. The probability of an event requiring initiation of the protective function within the proposed AOT is low. Additionally, surveillance testing is not generally performed on multiple sensors simultaneously. So, other trip functions and sensors remain operable and the probability of extensive inoperabilities affecting diverse trip functions is low. A spurious trip of a coincident channel could initiate a plant transient (for example, a reactor scram or a main steam isolation valve closure); however, these transients are bounded by the current analyses. Moreover, the original TS bases submitted as part of the application for Millstone Unit No. 1's Provisional Operating License (dated October 7, 1970) included recognition that instruments would be inoperable during required functional tests and calibrations. Thus, these conditions were recognized in the original design bases and constitute part of the licensing bases of the plant. NUREG-0123 provided specific time frames[,] ...AOTs addressed in the table notes[,] and specific action statements. Millstone Unit No. 1 AOT values chosen are consistent with these values and less than those approved in NUREG-1433 which had a more detailed study performed to lengthen the AOT value.

The existing TS definition for Instrument Functional Test would be difficult to satisfy if the LCO condition of tripping the inoperable channel was performed. A similar problem of complying with the Instrument Calibration definition also exists. The TS requirement to perform functional tests and calibrations is not consistent with a requirement to trip the system under test. The proposed TS changes permit more complete functional and calibration testing. For example, the main scram contactors could be included within the surveillance tests. Therefore, these TS clarifications do not increase the consequences of any previously analyzed accidents.

The two-hour instrumentation AOT for the Air Ejector Off-Gas System radiation monitors is slightly less restrictive than that allowed by the NUREG-0123. Since this requirement was relocated from NUREG-1433, there is no corresponding requirement for comparison. These radiation monitors are arranged in a two-out-of-two logic; therefore, both must trip to initiate the required action (closure of the off-gas isolation valve to the main stack). This action, however, is automatically delayed by 15 minutes. A high radiation condition sensed by the monitor in service would provide sufficient time to take corrective actions. Since a two-hour AOT is deemed acceptable for instrumentation in system[s] such as the Reactor Protection System and Emergency Core Cooling Systems, it is appropriate to apply a two-hour AOT to these radiation monitors. Additionally, the NUREG-0123 AOT of one hour does not allow sufficient time to perform required surveillance testing without placing undue stress on the test performer. The probability of a plant transient (e.g., loss of condenser vacuum) resulting from a trip of the coincident channel during surveillance testing is reduced since the channel under test will only be tripped for a small portion of the test interval. This transient is bounded by existing analyses. Therefore, this proposed AOT will not significantly increase the probability or consequences of an accident previously evaluated.

Since no physical change is being made to the secondary containment, or to any systems or components that interface with the secondary containment, there is no change in the probability of any accident analyzed in the UFSAR [Updated Final Safety Analysis Report].

The proposed change continues to ensure the secondary containment requirements meet the licensing basis. Also, the proposed changes are based on Standard Technical Specifications, NUREG-1433, ''Standard Technical Specifications General Electric Plants, BWR/4,'' Revision 1 guidelines and implement actions to be taken when secondary containment integrity is not met. If secondary containment integrity is not met, existing TS 3.7.C directs the plant to be placed in an operating condition where secondary containment is not required, e.g., COLD SHUTDOWN. A four hour allowable outage time is proposed which provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary containment during RUN, STARTUP/HOT STANDBY or HOT SHUTDOWN. The secondary containment is not an initiator for any accident. Therefore, the proposed change will not increase the probability of any previously analyzed accident. This short time period ensures that the probability of an accident requiring secondary containment integrity operability occurring during periods when secondary containment integrity is inoperable is minimal.

The proposed surveillance requirement is based on the NUREG-1433 surveillance requiring periodic confirmation that at least one door in each of the double-doored accesses to the secondary containment is closed, provides additional assurance of secondary containment system integrity. While this is a deviation from NUREG-1433 (which requires that both doors in each access be closed except for normal entry and exit), it is consistent with the current definition of Secondary Containment Integrity, which requires that at least one door in each access opening be closed. Hence, the deviation is justifiable and represents increased passive testing which will provide increased awareness of plant conditions. Increased awareness of plant conditions should reduce the probability or consequences of any accident previously evaluated.

Since the aspects of secondary containment integrity affected by reactor building access control are being revised in this proposed amendment to agree with the allowable outage time allowed by NUREG-1433 upon loss of secondary containment integrity, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Since the editorial items do not alter the meaning or intent of any requirements, they do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Millstone Nuclear Power Station, Unit No. 1, 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 to the protective instrumentation trip system specifications do not create the possibility of a new or different kind of accident because they do not introduce any new operational modes or physical modifications to the plant.

Instruments addressed in the proposed TS respond to changes in the plant. The proposed AOTs provide a two-hour interval where the instrument is inoperable, yet the TS LCO action statement is not immediately entered. Given a single failure, this could impact the response of the trip channel but not the initiation of the event. The only action resulting from the AOTs is to perform testing as required by TS. Spurious signals during testing could initiate transients but would be bounded by the previous transient analyses. These tests do not subject the instruments to any conditions beyond their design specifications and are performed in accordance with approved testing standards. This testing ensures equipment operability by identifying degraded conditions, initiating corrective action and properly retesting them. Therefore, the proposed TS changes will not introduce a new or different kind of accident than previously evaluated.

The two-hour instrumentation AOT for the Air Ejector Off-Gas System radiation monitors is slightly less restrictive than that allowed by the NUREG-0123. Since this requirement was relocated from NUREG-1433, there is no corresponding requirement for comparison. These radiation monitors are arranged in a two-out-of-two logic; therefore, both must trip to initiate the required action (closure of the off-gas isolation valve to the main stack). This action, however, is automatically delayed by 15 minutes. A high radiation condition sensed by the monitor in service would provide sufficient time to take corrective actions. Since a two-hour AOT is deemed acceptable for instrumentation in system[s] such as the Reactor Protection System and Emergency Core Cooling Systems, it is appropriate to apply a two-hour AOT to these radiation monitors.

The proposed changes to Millstone Unit No. 1 Technical Specifications Section 3.7/4.7 and associated bases were developed using the guidance provided in the Standard Technical Specifications, NUREG-1433, ''Standard Technical Specifications General Electric Plants, BWR/4,'' Revision 1. Augmentation of the existing surveillance requirements by incorporation of an additional NUREG-1433 based surveillance, provides additional assurance of secondary containment system integrity. While this is a deviation from NUREG-1433 (which requires that both doors in each access be closed except for normal entry and exit), it is consistent with the current definition of Secondary Containment Integrity which requires that at least one door in each access opening be closed. Hence, the deviation is justifiable and represents increased passive testing which will provide increased awareness of plant conditions. Increased awareness of plant conditions will not create the possibility of a new or different kind of accident from any accident previously evaluated. Since the proposed changes do not significantly degrade the present level of system operability and add provisions from NUREG-1433, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Since the editorial items do not alter plant configurations or operating modes, they do not create the possibility of a new or different kind of accident.

3. The operation of Millstone Nuclear Power Station, Unit No. 1, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The protective instrumentation surveillance requirements provide verification of the operability of the trip system instrumentation channels. In addition, the channel that monitors the identical Trip Function within the same trip system maintains trip capability for the relatively short duration that the coincidence change is in effect. This ensures that protective instrumentation reliability is maintained. The proposed change provides for a specific time period to perform required surveillances on instrument channels without trips present in associated trip systems. This time allotment tends to enhance the margin of safety by decreasing the probability of unnecessary challenges to safety systems and inadvertent plant transients.

The proposed TS provide a two-hour interval where the instrument is inoperable, yet the TS LCO action statement is not immediately entered. If a single failure occurred on the other channel of the trip system being tested and the channel being tested was not in the tripped condition, a valid signal might not provide the required protective action. The probability of an event requiring initiation of the protective function within the proposed AOT is low. Additionally, surveillance testing is not generally performed on multiple sensors simultaneously. So, other trip functions and sensors remain operable and the probability of extensive inoperabilities affecting diverse trip functions is low.

The existing TS definition for Instrument Functional Test would be difficult to satisfy if the LCO condition of tripping the inoperable channel was performed. A similar problem of complying with the Instrument Calibration definition also exists. Moreover, the original TS bases submitted as part of the application for Millstone Unit No. 1—s Provisional Operating License (dated October 7, 1970) included recognition that instruments would be inoperable during required functional test and calibrations. Thus, these conditions were recognized in the original design bases and constitute part of the licensing bases of the plant. NUREG-0123 provided specific time frames[,]...AOTs addressed in the table notes[,] and specific action statements. Millstone Unit No. 1 AOT values chosen are consistent with these values and less than those approved in NUREG-1433 which had a more detailed study performed to lengthen the AOT value.

The only action resulting from the proposed TS is to perform testing as required by TS. Spurious signals during testing could initiate equipment or plant transients but would be bounded by the previous transient analysis. These tests do not subject the instruments to any conditions beyond their design specifications and are performed in accordance with approved testing standards. This testing ensures equipment operability by identifying degraded conditions, initiating corrective action and properly retesting them. Therefore, the proposed TS do not involve a significant reduction in a margin of safety.

The two-hour instrumentation AOT for the Air Ejector Off-Gas System radiation monitors is slightly less restrictive than that allowed by the NUREG-0123. Since this requirement was relocated from NUREG-1433, there is no corresponding requirement for comparison. These radiation monitors are arranged in a two-out-of-two logic; therefore, both must trip to initiate the required action (closure of the off-gas isolation valve to the main stack). This action, however, is automatically delayed by 15 minutes. A high radiation condition sensed by the monitor in service would provide sufficient time to take corrective actions. Since a two-hour AOT is deemed acceptable for instrumentation in system[s] such as the Reactor Protection System and Emergency Core Cooling Systems, it is appropriate to apply a two-hour AOT to these radiation monitors and does not involve a significant reduction in the margin of safety.

The addition of an allowable outage time of four hours for Secondary Containment Integrity has negligible effect on accident occurrence or consequences. Since the proposed change does not involve the addition or modification of plant equipment, is consistent with the intent of the existing Technical Specifications, is consistent with the current industry practices as outlined in NUREG-1433, (except for the deviation noted above), and is consistent with the design basis of the plant and the accident analysis, no action will occur that will involve a significant reduction in a margin of safety.

Since the editorial items do not alter the meaning or intent of any requirements, they do not affect the margin of safety.

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

Local Public Document Room location: 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

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

NRC Deputy Director: Phillip F. McKee

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

Date of amendment requests: July 28, 1995, as revised February 21, 1997

Description of amendment requests: The proposed amendments would revise the Technical Specifications (TSs) to allow use of credit for soluble boron in spent fuel pool criticality analyses. The licensee's February 21, 1997, submittal is a revision to its original amendment requests dated July 28, 1995. The generic methodology for crediting soluble boron in spent fuel rack criticality analyses was approved by the NRC on October 25, 1996. However, because of changes made to the generic methodology as a result of comments from the NRC staff, it was necessary for NSP to revise its original amendment requests. In addition, the licensee has revised its request by eliminating the proposed relocation of the spent fuel pool operating limits to the Unit 1 core operating limits report and will retain these limits in the TSs.

The licensee's original application for amendments was published in the **Federal Register** on September 23, 1996, (61 FR 49800).

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

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

There is no increase in the probability of a fuel assembly drop accident in the spent fuel pool when considering the presence of soluble boron in the spent fuel pool water for criticality control. The handling of the fuel assemblies in the spent fuel pool has always been performed in borated water.

The criticality analysis showed the consequences of a fuel assembly drop accident in the spent fuel pool are not affected when considering the presence of soluble boron.

There is no increase in the probability of the accidental misloading of spent fuel assemblies into the spent fuel pool racks when considering the presence of soluble boron in the pool water for criticality control. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures and will be in accordance with the Technical Specification spent fuel rack storage configuration limitations. The addition of the spent fuel pool storage configuration surveillance in proposed Specification 4.20 will provide increased assurance that a spent fuel pool inventory verification will be completed in a timely manner after completion of a fuel handling campaign in the spent fuel pool.

There is no increase in the consequences of the accidental misloading of spent fuel assemblies into the spent fuel pool racks because criticality analyses demonstrate that the pool will remain subcritical following an accidental misloading if the pool contains an adequate boron concentration. The proposed Technical Specifications limitations will ensure that an adequate spent fuel pool boron concentration will be maintained.

There is no increase in the probability of the loss of normal cooling to the spent fuel pool water when considering the presence of soluble boron in the pool water for subcriticality control since a high concentration of soluble boron has always been maintained in the spent fuel pool water.

A loss of normal cooling to the spent fuel pool water causes an increase in the temperature of the water passing through the stored fuel assemblies. This causes a decrease in water density which would result in a decrease in reactivity when Boraflex neutron absorber panels are present in the racks. However, since Boraflex is not considered to be present, and the spent fuel pool water has a high concentration of boron, a density decrease causes a positive reactivity addition. However, the additional negative reactivity provided by the proposed 1800 ppm boron concentration limit, above that provided by the concentration required to maintain K_{eff} less than or equal to 0.95 (750 ppm), will compensate for the increased reactivity which could result from a loss of spent fuel pool cooling event. Because adequate soluble boron will be maintained in the spent fuel pool water, the consequences of a loss of normal cooling to the spent fuel pool will not be increased.

Therefore, based on the conclusions of the above analysis, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment[s] will not create the possibility of a new or different kind of accident from any accident previously analyzed.

Spent fuel handling accidents are not new or different types of accidents, they have been analyzed in Section 14.5.1 of the Updated Safety Analysis Report.

Criticality accidents in the spent fuel pool are not new or different types of accidents, they have been analyzed in the Updated Safety Analysis Report and in Criticality Analysis reports associated with specific licensing amendments for fuel enrichments up to 5.0 weight percent U-235.

The Prairie Island Technical Specifications currently contain limitations on the spent fuel pool boron concentration. Current Specification 3.8.E.2, which covers the storage of restricted fuel assemblies in an unverified condition, and Specification 3.8.B.1.c for the loading of fuel assemblies into a cask in the spent fuel pool, contain requirements for spent fuel pool boron concentration. The actual boron concentration in the spent fuel pool has always been kept at a higher value for refueling purposes. New Specification 3.8.E.2 establishes new boron concentration requirements for the spent fuel pool water consistent with the results of the new criticality analysis (Exhibit E [of the February 21, 1997, submittal]).

Since soluble boron has always been maintained in the spent fuel pool water, and is currently required by Technical Specifications under some circumstances, the implementation of this new requirement will have little effect on normal pool operations and maintenance. The implementation of the proposed new limitations on the spent fuel pool boron concentration will only result in increased sampling to verify boron concentration. This increased sampling will not create the possibility of a new or different kind of accident.

Because soluble boron has always been present in the spent fuel pool and is required by current Technical Specifications as discussed above, a dilution of the spent fuel pool soluble boron has always been a possibility. However, it was shown in the spent fuel pool dilution evaluation (Exhibit D [of the February 21, 1997, submittal]) that a dilution of the Prairie Island spent fuel pool which could reduce the rack K_{eff} to less than 0.95 is not a credible event. Therefore, the implementation of new limitations on the spent fuel pool boron concentration will not result in the possibility of a new kind of accident.

Revised Specifications 3.8.E.1, 5.6.A.1.d and 5.6.A.1.e continue to specify the requirements for the spent fuel rack storage configurations, the only significant changes relate to the criteria for determining the storage configuration. Since the proposed spent fuel pool storage configuration limitations will be similar to those currently in the Prairie Island Technical Specifications, the new limitations will not have any significant effect on normal spent fuel pool operations and maintenance and will not create any possibility of a new or different kind of accident. Verifications will continue to be performed to ensure that the spent fuel pool loading configuration meets specified requirements.

As discussed above, the proposed changes will not create the possibility of a new or different kind of accident. There is no significant change in plant configuration, equipment design or equipment. The accident analysis in the Updated Safety Analysis Report remains bounding.

3. The proposed amendment[s] will not involve a significant reduction in the margin of safety.

The Technical Specification changes proposed by this License Amendment Request and the resulting spent fuel storage operating limits will provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. Those limits are based on a plant specific criticality analysis (Exhibit E) performed in accordance [with] the Westinghouse spent fuel rack criticality analysis methodology described in Reference 4 [in Exhibit A of the February 21, 1997, submittal].

While the criticality analysis utilized credit for soluble boron, a storage configuration has been defined using a 95/95 K_{eff} calculation to ensure that the spent fuel rack K_{eff} will be less than 1.0 with no soluble boron. Soluble boron credit is used to offset uncertainties, tolerances and off-normal conditions and to provide subcritical margin such that the spent fuel pool K_{eff} is maintained less than or equal to 0.95.

The loss of substantial amounts of soluble boron from the spent fuel pool which could lead to exceeding a K_{eff} of 0.95 has been evaluated (Exhibit D) and shown to be not credible.

The evaluations in Exhibit D, which show that the dilution of the spent fuel pool boron concentration from 1800 ppm to 750 ppm is not credible, combined with the 95/95 calculation, which shows that the spent fuel rack K_{eff} will remain less than 1.0 when flooded with unborated water, provide a level of safety comparable to the conservative criticality analysis methodology required by References 1, 2 and 3 [in Exhibit A of the February 21, 1997, submittal].

Therefore, the proposed changes in this license amendment will not result in a significant reduction in the plant's margin of safety.

Based on the evaluation above, and pursuant to 10 CFR 50, Section 50.91, Northern States Power Company has determined that operation of the Prairie Island Nuclear Generating Plant in accordance with the proposed license amendment request does not involve any significant hazards considerations as defined by NRC regulations in 10 CFR 50, Section 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 requests involve no significant hazards consideration.

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

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

NRC Project Director: John N. Hannon

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

Date of amendment requests: February 14, 1997

Description of amendment requests: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant (DCPP) Unit Nos. 1 and 2 to revise the surveillance frequencies from at least once every 18 months to at least once per refueling interval (nominally 24 months) for 8 slave relay tests, 20 electrical system tests and 1 electrical TS Bases change, and 5 miscellaneous tests.

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 proposed TS surveillance interval increase to 24 months do not alter the intent or method by which the inspections, tests, or verifications are conducted; do not alter the way any structure, system, or component functions; and do not change the manner in which the plant is operated.

The surveillance, maintenance, and operating histories indicate that the equipment will continue to perform satisfactorily with longer surveillance intervals. Few surveillance and maintenance problems were identified. No problems have recurred following identification of root causes and implementation of corrective actions.

There are no known mechanisms that would significantly degrade the performance of the evaluated equipment during normal plant operation. All potential time related degradation mechanisms have insignificant effects in the timeframe of interest (24 months +25 percent, or 30 months). Based on the past performance of the equipment, the probability or consequences of accidents would not be significantly affected by the proposed surveillance interval increases.

Deletion of the phrase ''during shutdown'' for the applicable electrical TS will not alter the intent or method by which the inspections, tests, or verifications are conducted; nor alter the way any structure, system, or component functions. DCPP has administrative programs in place which require evaluation of risk and suitability of surveillance and maintenance activities to ensure that performance during plant operation does not adversely affect safety.

The administrative change for one PORV TS regarding channel calibration only maintains the existing surveillance frequency. This revision does not alter the intent or method by which the inspections, tests, or verifications are conducted; nor alter the way any structure, system, or component functions.

Therefore, the proposed changes do 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.

For the proposed TS changes involving surveillance interval increases to 24 months, the surveillance and maintenance histories indicate that the equipment will continue to effectively perform its design function over the longer operating cycles. Additionally, the increased surveillance intervals do not result in any physical modifications, affect safety function performance or the manner in which the plant is operated, or alter the intent or method by which surveillance tests are performed. No problems have recurred following identification of root causes and implementation of corrective actions. All identified potential time related degradations have insignificant effects in the timeframe of interest. The proposed surveillance interval increases would not affect the type of accident possible.

Deletion of the phrase —during shutdown— for the applicable electrical TS does not result in any physical modifications, affect safety function performance or the manner in which the plant is operated, or alter the intent or method by which surveillance tests are performed. DCPP has administrative programs in place which require evaluation of risk and suitability of surveillance and maintenance activities to ensure that performance during plant operation does not adversely affect safety.

The administrative change for one PORV TS regarding channel calibration only maintains the existing surveillance frequency. This revision does not result in any physical modifications, affect safety performance or the manner in which the plant is operated, or alter the intent or method by which surveillance tests are performed.

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

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

For the proposed TS changes involving surveillance interval increases to 24 months, evaluation of historical surveillance and maintenance data indicates there have been few problems experienced with the evaluated equipment. There are no indications that potential problems would be cycle ength dependent or that potential degradation would be significant for the timeframe of interest; therefore, increasing the surveillance interval will have little, if any, impact on safety. There is no safety analysis impact since these changes will have no effect on any safety limit, protection system setpoint, or limiting condition for operation, and there are no hardware changes that would impact existing safety analysis acceptance criteria. Safety margins would not be significantly affected by the proposed surveillance interval increases.

Deletion of the phrase ''during shutdown'' for the applicable electrical TS has no safety analysis impact since these changes will have no effect on any safety limit, protection system setpoint, or limiting condition for operation, and there are no hardware changes that would impact existing safety analysis acceptance criteria. DCPP has administrative programs in place which require evaluation of risk and suitability of surveillance and maintenance activities to ensure that performance during plant operation does not adversely affect safety.

The administrative change for one PORV TS regarding channel calibration only maintains the existing surveillance frequency. This revision has no safety analysis impact.

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 requests involve no significant hazards consideration.

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

Attorney for licensee: Christopher J. Warner, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120

NRC Project Director: William H. Bateman

Portland General Electric Company, et al., Docket No. 50-344, Trojan Nuclear Plant, Columbia County, Oregon

Date of amendment request: January 16, 1997, as supplemented February 24, 1997.

Description of amendment request: The proposed amendment would allow pre-operational testing and load handling of spent fuel transfer and storage casks in the Trojan Fuel Building.

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

The proposed changes would not involve a significant increase in the probability or consequences of an accident previously evaluated. With the permanent cessation of operations at the Trojan Plant, the number of potential accidents was reduced to those types of accidents associated with the storage of irradiated fuel and radioactive waste storage and handling. Additional events were postulated for decommissioning activities due to the difference in the types of activities that were to be performed. The postulated accidents described in the Defueled Safety Analysis Report (DSAR) are generally classified as: 1) radioactive release from a subsystem or component, 2) fuel handling accident, and 3) loss of spent fuel decay heat removal capability. The postulated events described in the Decommissioning Plan are grouped as: 1) decontamination, dismantlement, and materials handling events, 2) loss of support systems (offsite power, cooling water, and compressed air), 3) fire and explosions, and 4) external events (earthquake, external flooding, tornadoes, extreme winds, volcanoes, lightning, toxic chemical release). These types of accidents are discussed below.

Radioactive release from a subsystem or component involves failure of a radioactive waste gas decay tank (WGDT) or failure of a chemical and volume control system holdup tank (HUT). For a failure of a WGDT, the radioactive contents are assumed to be principally noble gases krypton and xenon, the particulate daughters of some of the krypton and xenon isotopes, and trace quantities of halogens. For the failure of a HUT, the assumptions were full power operation with 1-percent failed fuel, 40 weeks elapsed since power operation, and 60,000 gallons of 120°F liquid released over a 2-hour period. However, the WGDTs and HUTs are no longer active and have been drained. Therefore, pre-operational testing and load handling activities cannot increase the probability of occurrence of a failure of a WGDT or HUT. Since the failure of a WGDT or HUT is no longer credible, the consequences of failure of a WGDT or HUT cannot significantly increase as a result of pre-operational testing and load handling.

The fuel handling accident involves a stuck or dropped fuel assembly that results in damage of the cladding of the fuel rods in one assembly and the release of gaseous fission products. Pre-operational testing and load handling do not involve the movement of irradiated fuel. A dummy assembly will be used for fit-up testing. The fuel handling equipment will be the same as previously analyzed with the exception of special tools that may be used to manipulate the dummy fuel assembly. These special tools will be similar in size and weight to other tools used for underwater manipulation, and therefore, would not present a new hazard. In addition, the same administrative controls and physical limitations imposed on any fuel handling operation will be used for pre-operational testing and load handling. Thus, there is no increase in the probability of occurrence of a fuel handling accident over what would be expected for any routine fuel handling operation. If a dummy fuel assembly were dropped in the spent fuel pool, then only one fuel assembly could be damaged. Therefore, the consequences of a dummy fuel assembly drop would be the same as the consequences of the analysis described in the DSAR. Therefore, the consequences of a dummy fuel assembly drop are not significantly increased as a result of pre-operational testing and load handling.

The loss of spent fuel decay heat removal capability involves the loss of forced spent fuel cooling with and without concurrent spent fuel pool (SFP) inventory loss. The only requirement to assume adequate decay heat removal capability for the spent fuel is to maintain the water level in the SFP so that the spent fuel assemblies remain covered (i.e., the capability to makeup water to the SFP must be available when required). The potential events that could result in a loss of spent fuel decay heat removal capability include external events (explosions, toxic chemicals, fires, ship collision with the intake structure, oil or corrosive liquid spills in the river, cooling tower collapse, seismic events, severe meteorological events), and internal events, including SFP makeup water system malfunctions. Pre-operational testing and load handling will not require the use of explosive materials, toxic chemicals, or flammable materials. The probability of other external events (e.g., cooling tower collapse) would be unaffected by the pre-operational testing and load handling activities inside the fuel building. Pre-operational testing and load handling activities will not directly interface with the SFP makeup water systems, and therefore could not affect their probability of failure. The safe load path and handling height limitations will ensure that a load drop does not adversely affect the SFP or makeup water systems. Therefore, there is no significant increase in the probability of a loss of spent fuel decay heat removal capability. There are no credible adverse consequences of the loss of spent fuel decay heat removal as the DSAR demonstrates that adequate time is available to establish a source of makeup water to the SFP such that uncovering the fuel and an actual loss of spent fuel cooling is not credible. The postulated events that could affect the SFP (liner tear/breach and heavy load drop) do not have a significant adverse effect. In addition, establishment of the makeup water path and recovery of spent fuel cooling would not be affected because postulated off-normal events and accidents could not affect the capability to provide makeup water to the SFP by various water sources. Therefore, pre-operational testing and load handling cannot significantly increase the consequences of the loss of spent fuel decay heat removal.

The events postulated in the Decommissioning Plan are similar to the DSAR with the exception of decontamination, dismantlement, and materials handling events. Decontamination events involve gross liquid leakage from in-situ decontamination equipment or accidental spraying of liquids containing concentrated contamination. Dismantlement events include segmentation of components and structures, or removal of concrete by rock splitting, explosives, or electric and/or pneumatic hammers. Dismantlement events potentially result in airborne contamination. Materials handling events involve dropping contaminated components, concrete rubble, or filters or packages of particulate materials. Pre-operational testing and load handling activities are material handling activities and are therefore, within the bounds of the existing analysis. Therefore, the probability and consequences of decontamination, dismantlement, and materials handling events would not be significantly increased.

Based on the above, the pre-operational testing and load handling activities do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated. As described in the licensee's safety evaluation of the proposed pre-operational testing and load handling activities, no types of off-normal events/accidents were determined to have radiological consequences greater than currently evaluated in the DSAR and Decommissioning Plan.

The postulated dummy fuel assembly drop is considered the same type or kind of event as the previously analyzed fuel handling accident, mainly because the initiator for this postulated event is the same (i.e., a (non-specified) failure of the fuel handling equipment or the fuel handling bridge crane. During pre-operational testing and load handling, a dummy fuel assembly could be dropped in the SFP or the cask loading pit. As the cask loading pit is similar in construction to the SFP and the cask loading pit will be flooded with borated water of the same concentration as the SFP, the differences between the two events are negligible and the two events may be considered the same type or kind of accident. Therefore the dummy fuel assembly drop is not a new or different type or kind of accident.

The postulated transfer cask drop or mishandling event is similar to a materials handling event. Therefore, the consequences of a transfer cask drop or mishandling event would not represent a new or different type or kind of accident.

Based on the above, the pre-operational testing and load handling activities do not create the possibility of a new or different kind of accident.

The proposed changes do not involve a significant reduction in the margin of safety. The Trojan Permanently Defueled Technical Specifications (PDTS) contain four limiting conditions of operation that address SFP water level, SFP boron concentration, SFP temperature, and SFP load restrictions. These PDTS will remain in effect as long as spent fuel is stored in the SFP, which is in accordance with their applicability statements. The pre-operational testing and load handling activities will not affect these PDTS or their bases.

The cask loading pit (CLP) is immediately adjacent to the SFP. The gate between the CLP and the SFP may be opened to allow a dummy fuel assembly to moved from the spent fuel storage racks in the SFP to the basket in the CLP. Opening the gate will allow free exchange of water between the CLP and the SFP. The water in the CLP must be at essentially the same level, boron concentration, and temperature as the SFP prior to the first opening of the gate to ensure that the limited conditions of operation are continuously satisfied for the SFP. Therefore, the CLP will be initially filled to about the same level as the SFP with water that is about the same boron concentration and temperature as the SFP. With these precautions, the limiting conditions of operation for SFP level, boron concentration, and temperature will be continuously maintained and the margin of safety will be unaffected.

Pre-operational testing and load handling activities will involve lifting and moving heavy loads (e.g., transfer casks). Loads that will be carried over fuel in the SFP racks and the heights at which they may be carried will be limited in accordance with LCO 3.1.4, ''Spent Fuel Pool Load Restrictions,'' in such a way as to preclude impact energies over 240,000 in-lbs. With this precaution, the limiting condition of operation pertaining to load restrictions over the SFP will be satisfied for fuel stored in the SFP racks and the margin of safety will be unaffected. The safe load path for heavy loads being lifted and moved outside the SFP will be located sufficiently far from the SFP as to not have an adverse effect on the SFP in the unlikely event of a load drop. In addition, the mechanical stops and electrical interlocks on the fuel building overhead crane will provide additional assurance that heavy loads are not carried over the fuel in the SFP racks.

Based on the above, the pre-operational testing and load handling activities will not reduce the margin of safety.

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: Branford Price Millar Library, Portland State University, 934 S.W. Harrison Street, P.O. Box 1151, Portland, Oregon 97207

Basis for proposed no significant hazards consideration determination:

Attorney for licensees: Leonard A. Girard, Esq., Portland General Electric Company, 121 S.W. Salmon Street, Portland, Oregon 97204

NRR Project Director: Seymour H. Weiss

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: January 2, 1997

Description of amendment request: The proposed amendment would allow a change to the current functional testing frequency for Inservice Inspection of American Society of Mechanical Engineers Code Class 1, 2, and 3 pumps and valves from the current monthly to a quarterly 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 the proposed license amendment involve a significant increase in the probability or consequences of an accident previously analyzed?

Response: Operation of Indian Point 3 in accordance with the proposed license does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes involve no hardware changes, no changes to the operation of any systems or components, and no changes to existing structures. 10 CFR 50.55a(g) requires that safety related components (e.g. - pumps and valves) be tested according to the requirements of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) and applicable addenda. The revision of functional test frequencies for pumps and valves, which are categorized as Code Class 1, 2, or 3, from a monthly to a quarterly test interval is consistent with NRC guidance provided in NUREG-1366 and in accordance with recommended test intervals in the ASME Code. These changes will reduce component degradation resulting from unnecessary tests and provide better system availability from not having to remove a system/component from operability while performing a surveillance. Such changes will not alter the probability or consequences of any previously analyzed accidents.

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

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

The proposed changes are procedural in nature concerning the functional testing frequencies of pumps and valves that have historically shown a high percentage of successfully meeting surveillance requirements. The methodology of testing these pumps and valves will remain unchanged. The proposed changes, while slightly increasing the possibility of an undetected pump or valve defect, will not create a new or unevaluated accident or operating condition.

(3) Does the proposed license 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 changes are in accordance with recommendations provided by the NRC regarding the improvement of Technical Specifications. These changes will result in the perpetuation of current safety margins while reducing the testing burden and decreasing equipment degradation.

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: S. Singh Bajwa, Acting Director

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

Date of amendment request: February 11, 1997

Description of amendment request: The proposed change to Hope Creek Technical Specification (TS) Sections 3/4.8.1 ''A.C. Sources,'' 6.8 ''Procedures and Programs,'' and the Bases for Section 3/4.8, ''Electrical Power Systems,'' would include: 1) the relocation of existing surveillance requirements related to diesel fuel oil chemistry; 2) the introduction of a new program under TS 6.8.4.e, ''Diesel Fuel Oil Testing Program;'' 3) revisions to the TS Bases for Section 3/4.8 to incorporate information associated with the TS changes; and 4) editorial changes to implement required corrections.

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

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

The proposed changes involve: 1) no hardware changes; 2) no significant changes to the operation of any systems or components in normal or accident operating conditions; and 3) no changes to existing structures, systems or components. Therefore these changes will not increase the probability of an accident previously evaluated.

Establishment of [Emergency Diesel Generator] EDG fuel oil testing requirements in TS 6.8.4.e is a change that is consistent with changes made in the improved STS [Standard Technical Specifications] as contained in Specification 5.5.10 of that document. These changes establish a new requirement to test for particulates in the EDG fuel oil, but establish a 92 day test frequency (as opposed to 31 days in the improved STS) and a 3.0 micron acceptance criteria (as opposed to 0.8 micron in the improved STS) for particulate testing. [Public Service Electric and Gas Company] PSE&G concludes that these changes are acceptable based upon past EDG fuel oil tests for particulates and acceptable performance of the EDG with 5.0 micron filters. In addition, PSE&G will utilize more objective test criteria for water and sediment in the EDG fuel oil than established by the ''clear and bright'' acceptance criteria contained in the improved STS.

Since the EDG fuel oil will still: 1) meet all of the requirements established for fuel oil specified in the improved STS; and 2) retain the capability to mitigate the consequences of accidents described in the [Hope Creek Generating Station] HC Safety Analysis Report, the proposed changes were determined to be justified. Based on established fuel oil quality history, the proposed testing methods and frequencies will not significantly decrease confidence in fuel oil quality and EDG operability, nor will they have any negative effect on established plant practices in regards to the testing of EDG fuel oil. Therefore, these changes will not involve a significant increase in the consequences of an accident previously evaluated.

The revisions proposed to the TS Bases are being made to provide additional information supporting the proposed EDG TS. With the approval of the proposed TS changes, the associated Bases changes would be editorial in nature. Therefore, these changes will not involve a significant increase in the consequences of an accident previously evaluated.

In addition, the proposed change to [Limiting Condition for Operation] LCO 3.8.1.1, ACTION c., is considered to be editorial in nature and will not result in a significant increase in the 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 HC EDGs are designed to mitigate the consequences of accidents by providing electrical power to safety-related equipment. Failure of the EDGs are not considered to initiate any of the accidents described in the HC Safety Analysis Report. The proposed changes concern fuel oil system surveillances and testing frequency. The proposed changes will not adversely impact the operation of any safety related component or equipment. Since the proposed changes involve: 1) no hardware changes; 2) no significant changes to the operation of any systems or components; and 3) no changes to existing structures, systems or components, there can be no impact on the occurrence of any accident. Furthermore, there is no change in plant testing proposed in this change request which could initiate an event. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

In addition, the proposed change to LCO 3.8.1.1, ACTION c., is considered to be editorial in nature and will not result in 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.

Establishment of EDG fuel oil testing requirements in TS 6.8.4.e is a change that is consistent with changes made in the improved STS. The proposed changes address: 1) how EDG fuel oil quality is to be determined; 2) how frequently this determination is to be performed; and 3) how to control the process for determining fuel oil acceptability and resultant EDG operability. With the exception of particulate testing (which is being added) all acceptance criteria for fuel oil testing remain unchanged. Based on historical data, EDG fuel oil quality will not be adversely affected or impacted by the proposed changes. Therefore, the proposed amendment does not involve any significant reduction in a safety margin.

The revisions proposed to the TS Bases are being made to provide additional information supporting the proposed EDG TS. With the approval of the proposed TS changes, the associated Bases changes would be editorial in nature. Therefore, these changes will not involve a significant reduction in a safety margin.

In addition, the proposed change to LCO 3.8.1.1, ACTION c., is considered to be editorial in nature and will not involve a significant reduction in a safety margin.

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: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

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

NRC Project Director: John F. Stolz

Tennessee Valley Authority, Docket No. 50-390 Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of amendment request: October 23, 1996, January 31, February 10 and 24 and March 11, 1997.

Description of amendment request: The proposed amendment would revise the Watts Bar Nuclear Plant (WBN) Unit 1 Technical Specifications to increase the enrichment and storage capacity of the spent fuel pool racks. The proposed modification increases the (Watts Bar Nuclear Plant) WBN spent fuel storage capacity from 484 fuel assemblies to 1835 fuel assemblies. The initial enrichment of the fuel to be stored in the spent fuel storage racks will be increased from 3.5 weight percent (wt%) to 5.0 wt%. This modification would also change the spacing of stored fuel assembly center-to-center spacing from a nominal 10.72 inches to 10.375 inches in 24 PaR flux trap rack modules and 8.972 inches in ten smaller burnup credit rack modules to be installed peripherally along the south and west pool walls and in a single 15 x 15 burnup credit rack to be installed in the cask pit.

In addition to the above proposed revisions, two limiting conditions for operation will be added to require that the combination of initial enrichment and burnup of each spent fuel assembly to be stored is in the acceptable region and to require boron concentration of the cask pit to be greater than or equal to 2000 parts per million (ppm) during fuel movement in the flooded cask pit. As an added protection to the fuel stored in the cask pit area, the Technical Requirements Manual (TRM) is being revised to require that an impact shield be in place over the fuel when heavy loads are moved near or across the cask pit area.

The WBN Unit 1 Technical Specification Bases and the TRM would be revised to support these 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:

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

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

The following potential scenarios were considered:

1. A spent fuel assembly drop.
2. Drop of the transfer canal gate or the cask pit divider gate.
3. A seismic event.
4. Loss-of-cooling flow in the spent fuel pool.
5. Installation activities.

The effect of additional spent fuel pool storage cells fully loaded with fuel on the first four potential accident scenarios listed above has been considered. It was concluded that after installation activities have been completed, the presence of additional fuel in the pool does not increase the probability of occurrence of these four events. Also, based on evaluations of bulk pool temperature, rack seismic responses, and refueling accidents, it is reasonable to conclude that there is no significant increase in the consequences of these events after installation is complete (See Reference 1). During the installation activities, the following considerations support a conclusion that neither the probability or consequences of these four scenarios would be significantly increased.

A spent fuel assembly cannot be dropped during installation of the 24 Programmed and Remote System Corporation (PaR) flux trap rack modules because this activity will take place before the end of operating cycle one and there will be no spent fuel in the WBN pool to be moved or shuffled. Before installing the ten smaller burnup credit racks in the pool, some fuel will be moved to create a three foot lateral free zone clearance from stored fuel. This would involve a one-time movement of an estimated maximum of 225 fuel assemblies, which is less that half the fuel movements during one refueling outage. This does not significantly increase the probability of dropping a fuel assembly, particularly when the many administrative controls and physical limitations imposed on fuel handling operations are considered. The fuel handling system consists of equipment and structures utilized for safely implementing refueling operations in accordance with requirements of General Design Criteria 61 and 62 of 10 CFR 50, Appendix A. The radiological dose consequences of dropping a 5.0 wt% fuel assembly are different from the previous FSAR [Final Safety Analysis Report] evaluation for the 3.5 wt% fuel assembly. The Beta and Gamma doses decrease and the maximum thyroid dose increase is less than 9%. Therefore, the change in calculated dose values is insignificant and remains well within regulatory guidelines.

It may be necessary to move the transfer canal gate and the cask pit divider gate between their gated and stored positions during installation of the burnup credit ''baby'' rack modules along the south and west walls. During rack installation, the previously mentioned three foot lateral free zone clearance to stored fuel would exist. Therefore, no heavy load would be carried directly over irradiated fuel during installation of the racks. There are numerous design features which comply with NUREG-0612 to preclude these gates from dropping on spent fuel. These features include design of the lifting devices, design of the crane, and use of written procedures. Also, the evaluation results for a gate drop on the racks indicates that permanent damage to a fuel storage cell is limited to a maximum depth of less than six inches below the top of the rack with no effect on the subcriticality of fuel stored in adjacent cells. Based on the foregoing, it is reasonable to conclude that gate handling during the installation of the ''baby'' racks would not involve a significant increase in the probability or consequences of an accident.

The probability of a seismic event is not related to installation activities. The worst consequence resulting from a seismic event during installation activities would occur during handling of a rack. The consequences would be insignificant because the Auxiliary Building crane is seismically qualified and both handling equipment and operations meet the criteria of NUREG-0612. Nevertheless, if the seismic event resulted in a rack drop, the consequences are insignificant, i.e., localized damage to the pool liner and a minor leak rate which would be small in comparison to available installed makeup capacity. The cooling and shielding of the spent fuel would remain unaffected. Also the racks being moved are empty during installation and therefore, the criticality consequences of seismic events are bounded by evaluations for loaded racks.

Rack installation activities cannot cause an accidental loss-of-cooling flow in the spent fuel pool. The vital components of the spent fuel pool cooling and cleanup system (SFPCCS) are not located proximate to the pool installation activities. Coolant flow may be deliberately curtailed to facilitate installation of the ''baby'' racks directly beneath the discharge piping in the southwest corner of the pool. The effects of such an action would be readily minimized and made inconsequential during the detailed installation planning phase by selecting a time when decay heat input from stored fuel is relatively constant. Also careful preplanning of the work would minimize out-of-service time and provide for intermittent coolant flow restart, if necessary, to maintain acceptable bulk coolant temperatures. Similarly, the effect of an independently initiated loss-of-coolant flow incident on reracking activities can be easily accommodated by stopping work, as necessary, to mitigate any adverse effects on the installation process. The consequences of loss-of-cooling flow in the spent fuel pool during installation are bounded by the analysis in Chapter 5 of the report which includes the situation in which ''baby'' racks and the 15 x 15 cask pit rack are installed, and the pool is filled to capacity with spent fuel.

With regard to the actual installation activities, the existing WBN TRM prohibits loads in excess of 2059 pounds from travel over fuel assemblies in the storage pool and requires the associated crane interlocks and physical stops be periodically demonstrated operable. During installation, racks and associated handling tools will be moved over the spent fuel pool, however there will be no fuel in the pool when the 24 flux trap rack modules are installed. A three foot lateral free zone clearance from stored spent fuel will be maintained during installation of the ten smaller burnup credit rack modules. Installation work in the spent fuel pit area will be controlled and performed in strict accordance with specific written instructions.

NUREG-0612 states that in lieu of providing a single failure-proof crane system, the control-of-heavy-loads guidelines can be satisfied by establishing that the potential for a heavy load drop is extremely small. Storage rack movements to be accomplished with the WBN Auxiliary Building crane will conform with NUREG-0612 guidelines in that the probability of a drop of a storage rack is extremely small. The crane has a tested capacity of 125 tons. The maximum weight of any existing, replacement, or new storage rack and its associated handling tool is less than 20 tons. Therefore, there is ample safety factor margin for movements of the storage racks by the Auxiliary Building crane. Special lifting devices, which have redundancy or a rated capacity sufficient to maintain adequate safety factors, will also be utilized in the movements of the storage racks. In accordance with NUREG-0612, Appendix B, the safety margin ensures that the probability of a load drop is extremely low.

Future load travel over fuel stored in a rack specifically designed for the cask loading area of the cask pit will be prohibited unless an impact shield, which has been specifically designed for this purpose, is covering the area. Loads that are permitted when the shield is in place must meet analytically determined weight, travel height, and cross-sectional area criteria that preclude penetration of the shield. A Technical Requirement (TR) has been proposed that incorporates the previously mentioned load criteria.

Also a rack change-out sequence is being developed that addresses removal of the existing racks, movement of the new racks into the Auxiliary Building, initial staging on the refueling floor, and final installation in the pool. The change-out sequence objectives include establishing lift heights, travel distances, and number of lifts to be as low as reasonably achievable. Accordingly, it is concluded that the proposed installation activities will not significantly increase the probability of a load-handling accident. The consequences of a load-handling accident are unaffected by the proposed installation activities.

The consequences of a spent fuel assembly drop were evaluated, and it was determined that the racks will not be distorted such that the racks would not perform their safety function. The criticality acceptance criterion, K_{eff} less than or equal to 0.95, is not violated, and the calculated doses are well within 10 CFR Part 100 guidelines. The radiological consequences of the fuel assembly drop accident evaluated for WBN, have changed, however, the changes do not involve a significant increase in consequences and are well within the 10 CFR 100 requirements.

A TRM change has been proposed that would permit the transfer-canal gate and the divider gate for the cask pit to travel over fuel assemblies in the spent fuel pool during movement between their gated and stored position. Rack damage is restricted to an area above the active fuel region, therefore, neither criticality nor radiological concerns exist.

The consequences of a seismic event have been evaluated. The replacement racks are designed and fabricated and the new racks will be fabricated to meet the requirements of applicable portions of the NRC regulatory guides and published standards. Design margins have been provided for rack tilting, deflection, and movement such that the racks do not impact each other or the spent fuel pool walls in the active fuel region during the postulated seismic events. The free-standing racks will maintain their integrity during and after a seismic event. The fuel assemblies also remain intact and therefore no criticality concerns exist.

The spent fuel pool system is a passive system with the exception of the fuel pool cooling train and heating, ventilating, and air-conditioning (HVAC) equipment. Redundancies in the cooling train and HVAC hardware are not reduced by the planned fuel storage modification. The potential increased heat load resulting from any additional storage of spent fuel is well within the existing system cooling capacity. Therefore, the probability of occurrence or malfunction of safety equipment leading to the loss-of-cooling flow in the spent fuel pool is not significantly affected. Furthermore, the consequences of this type incident are not significantly increased from previously evaluated cooling system loss of flow malfunctions. Thermal-hydraulic scenarios assume the reracked pool is approximately 90% full with spent fuel assemblies. From this starting point, the remaining storage capacity is utilized by analyzing both normal and unplanned full core off loads using conservative assumptions and previously established methods. Calculated values include maximum pool water bulk temperature, coincident maximum pool water local temperature, the maximum fuel cladding temperature, time-to-boil after loss-of-cooling paths, and the effect of flow blockage in a storage cell.

Although the proposed modification increases the pool heat load, results from the above analyses yield a maximum bulk temperature less than 160 degrees Fahrenheit which is below the bulk boiling temperature. Also the maximum local water temperature is below nucleate boiling condition values. Associated results from corresponding loss-of-cooling evaluations give minimums of 5.3 hours before boiling begins and 45 hours before the pool water level drops to the minimum required for shielding spent fuel. This is sufficient time to begin utilization of available alternate sources of makeup cooling water. Also, the effect of the increased thermal loading on the pool structure, associated cooling system, and components was evaluated and determined to establish an acceptable design basis with the new storage configuration. No modifications were necessary because of the increased temperature.

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

The proposed modification has been evaluated in accordance with the guidance of the NRC position paper entitled, ''OT Position for Review and Acceptance of Spent-Fuel Storage and Handling Applications'', appropriate NRC regulatory guidelines; appropriate NRC standard review plans; and appropriate industry codes and standards. Proven analytical technology was used in designing the planned fuel storage expansion and will be utilized in the installation process. Basic reracking technology has been developed and demonstrated in applications for fuel pool capacity increases that have already received NRC staff approval.

Proposed TSs for the spent fuel storage racks use burnup credit and fuel assembly administrative placement restrictions for criticality control. These restrictions are described in the proposed change to the design features section of the TSs by reference to the Spent Fuel Pool Modifications report. Additional evaluations were required to ensure that the criticality criterion, k_{eff} less than 0.95, is maintained. These include evaluation for the abnormal placement of unirradiated (fresh) fuel assemblies of 5.0 wt% enrichment into a storage cell location designed for lower enrichment or irradiated fuel. Soluble boron, for which credit is permitted under these abnormal conditions, ensures that reactivity is maintained substantially less than the design requirement. For example, if the PaR flux trap racks are inadvertently all loaded with fresh assemblies of the maximum 5.0 wt% fuel instead of observing the 3.8 wt% and 6.75 MWD/KgU controls, the worth of the 2000 ppm borated water is sufficient to lower the k_{eff} of the storage racks to 0.83. The existing and proposed TSs require boron concentration in the pool and cask pit to be more than or equal to 2000 ppm during fuel movement. An analytical determination of the reactivity worth of 2000 ppm borated water in the spent fuel storage pool predicted the change in k_{eff} to be approximately 17 percent k_{eff}. Although no credit for soluble boron was proposed in the TSs, it was also determined by an independent calculation that a minimum concentration of 520 ppm soluble boron allows the unrestricted storage of 5.0 wt% enriched fuel in the PaR flux trap racks.

The Holtec-designed peripheral ''baby'' racks and the 15 x 15 racks in the cask loading area can safely and conservatively store fuel of 5 wt% initial enrichment burned to 41 MWD/kgU or lower enriched fuel with lower burnup, i.e., fuel of equivalent reactivity. Evaluations have confirmed that, for the abnormal placement of a fresh fuel assembly of 5.0 wt% in these racks, the criticality criterion is maintained with the existing and proposed TS requirements of 2000 ppm soluble boron.

Although these changes required addressing additional aspects of a previously analyzed accident, the possibility of a previously unanalyzed accident is not created.

The impact shield design together with its attendant administrative controls and NUREG-0612 heavy load lift compliance, renders the possibility of a heavy load drop on fuel as not credible in accordance with the NUREG-0612 single-failure-proof criteria. Accordingly, since this particular part of the proposed reracking modification is not a change that could malfunction by a new single failure, the movement of heavy loads over the cask pit does not create the possibility of a new or different kind of accident.

It is therefore concluded that the proposed reracking does not create the possibility of a new or different kind of accident from any previously analyzed.

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

The design and technical review process applied to the reracking modification included addressing the following areas:

1. Nuclear criticality considerations.

2. Thermal-hydraulic considerations.

3. Mechanical, material, and structural considerations.

The established acceptance criterion for criticality is that the neutron multiplication factor shall be less than or equal to 0.95, including all uncertainties. The results of the criticality analyses for the rack designs demonstrate that this criterion is satisfied. The methods used in the criticality analysis conform to the applicable portions of NRC guidance and industry codes, standards, and specifications. In meeting the acceptance criteria for criticality in the spent fuel pool and the cask loading area, such that k_{eff} is always less than 0.95 at a 95/95 percent probability tolerance level, the proposed amendment does not involve a significant reduction in the margin of safety for nuclear criticality.

Conservative methods and assumptions were used to calculate the maximum fuel temperature and the increase in temperature of the water in the spent fuel pit area. The thermal-hydraulic evaluation used methods previously employed. The proposed storage modification will increase the heat load in the spent fuel pool, but the evaluation shows that the existing spent fuel cooling system will maintain the bulk pool water temperature at or below 160 degrees Fahrenheit. Thus it is demonstrated that the worst-case peak value of the pool bulk temperature is considerably lower than the bulk boiling temperature. Evaluation also shows that maximum local water temperatures along the hottest fuel assembly are below the nucleate boiling condition value. Thus, there is no significant reduction in the margin of safety for thermal hydraulic or spent fuel cooling considerations.

The mechanical, material, and structural design of the spent fuel racks is in accordance with applicable portions of NRC—s position in ''OT Position for Review and Acceptance of Spent-Fuel Storage and Handling applications,'' dated April 14, 1978 (as modified January 18, 1979), as well as other applicable NRC guidance and industry codes. The primary safety function of the spent fuel racks is to maintain the fuel assemblies in a safe configuration through normal and abnormal loading conditions. Abnormal loadings that have been evaluated with acceptable results and discussed previously include the effect of an earthquake and the impact because of the drop of a fuel assembly. The rack materials used are compatible with the fuel assemblies and the environment in the spent fuel pool. The structural design for the new racks provides tilting, deflection, and movement margins such that the racks do not impact each other or the spent fuel pit walls in the active fuel region during the postulated seismic events. Also the spent fuel assemblies themselves remain intact and no criticality concerns exist. In addition, finite element analysis methods were used to evaluate the continued structural acceptability of the spent fuel pit. The analysis was performed in accordance with ''Building Code Requirements for Reinforced Concrete,'' (ACI 318-63,77). Therefore, with respect to mechanical, material, and structural considerations, there is no significant reduction in a margin of safety.

Summary

Based on the above analysis, TVA has determined that operation of WBN, in accordance with the proposed amendment, would not: (1) involve a significant increase in the probability of consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Therefore, operations of WBN in accordance with the proposed amendments as described do not involve significant hazard considerations as defined in 10 CFR 50.92 and that the criteria of 10 CFR 50.91 have accordingly been met.

TVA has also reviewed the NRC examples of licensing amendments considered not likely to involve significant hazards considerations as provided in the final adoption of 10 CFR 50.92 published on page 7751 of the Federal Register, Volume 51, No. 44, March 6, 1986. Example (X) provides four criteria that, if satisfied by a reracking request, indicate that it is likely no significant hazards considerations are involved. The criteria and how TVA—s amendment request for WBN complies are indicated below.

Criterion (1):

The storage expansion method consists of either replacing existing racks with a design that allows closer spacing between stored spent fuel assemblies or replacing additional racks of the original design on the pool floor if space permits.

Proposed Amendment:

The WBN reracking involves replacing the existing racks with a design that allows slightly closer spacing between stored fuel assemblies and also provides additional rack storage on the pool floor where space permits.

Criterion (2):

The storage expansion method does not involve rod consolidation or double tiering.

Proposed Amendment:

The WBN racks are not double tiered, and the racks will sit on the floor of the spent fuel pool. Additionally, the amendment application does not involve consolidation of spent fuel.

Criterion (3):

The k_{eff} of the pool is maintained less than or equal to 0.95.

Proposed Amendment

The design of the spent fuel racks contains a neutron absorber, Boral, to allow close storage of spent fuel assemblies while ensuring that the k_{eff} remains less than 0.95 under normal operating conditions with unborated water in the pool and less than 0.95 under abnormal conditions with soluble boron in the pool.

Criterion (4):

No new technology or unproven technology is utilized in either the construction process or the analytical techniques necessary to justify the expansion.

Proposed Amendment:

The construction processes and analytical techniques used in the fabrication and design are substantially the same as those of numerous other rack installations, Thus, no new or unproven technology is utilized in the construction or analysis of the high density, spent fuel racks at WBN. TVA's contractor, Holtec International, has previously supplied licensable racks of several similar design for about 10 other reracking projects

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: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, TN 37402

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

NRC Project Director: Frederick J. Hebdon

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: November 26, 1996

Description of amendment request: The proposed changes would eliminate the records retention requirements from the administrative section of the Technical Specifications (TS) in accordance with NRC Administrative Letter 95-06, ''Relocation of Technical Specifications Administrative Controls Related to Quality Assurance.''

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

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

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed administrative changes do not affect equipment or its operation. Therefore, the likelihood that an accident will occur is neither increase nor decreased by relocating record retention requirements from the Technical Specifications to the Operational Quality Assurance Program. This TS change will not impact the function or method of operation of plant equipment. Thus, a significant increase in the probability of a previously analyzed accident does not result due to this change. No systems, equipment, or components are affected by the proposed changes. Thus, the consequences of any accident previously evaluated in the UFSAR [Updated Final Safety Analysis Report] are not increased by this change.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not alter the design or operations of the physical plant. Since record retention requirements are administrative in nature, a change to these requirements does not contribute to accident initiation, an administrative change related to this activity does not produce a new accident scenario or produce a new type of equipment malfunction. [These] changes do not alter any existing accident scenarios. The proposed administrative change does not affect equipment or its operation, and, thus, does not create the possibility of a new or different kind of accident. Therefore, the proposed change does not create the possibility of a new or different kind of accident.

(3) Involve a significant reduction in a margin of safety. Section 6.0 of the...Surry Technical Specifications does not have a basis description. The proposed administrative change does not affect equipment or its operation, and, thus, does not involve any 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: 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: Mark Reinhart, Acting

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.

Baltimore Gas and Electric Company, Docket No. 50-318, Calvert Cliffs Nuclear Power Plant, Unit No. 2, Calvert County, Maryland

Date of application for amendment: July 31, 1997, as supplemented February 13, 1997.

Brief description of amendment: The proposed amendment would revise the Technical Specifications to reduce the minimum Reactor Coolant System total flow rate from 370,000 gpm to 340,000 gpm. The proposed changes are necessary to support a larger number of plugged steam generator tubes for future operating cycles.

Date of publication of individual notice in **Federal Register:** February 26, 1997 (62 FR 8780)

Expiration date of individual notice: March 28, 1997

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

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

Date of application for amendment: February 14, 1997

Brief description of amendment: The proposed amendment would revise the Technical Specifications to permit a one-time extension of the current steam generator tube inservice inspection cycle. Date of publication of individual notice in **Federal Register:** March 4, 1997 (62 FR 9816)

Expiration date of individual notice: March 28, 1997

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

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

Date of application for amendment: February 17, 1997

Brief description of amendment: Changes to Technical Specification to implement 10 CFR 50, Appendix J Option B relating to containment leakage tests.

Date of publication of individual notice in the **Federal Register:** February 28, 1997 (62 FR 9214).

Expiration date of individual notice: March 31, 1997

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 32629

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

Date of amendment request: February 14, 1997

Brief description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 3/4.5.2, ''Emergency Core Cooling Systems, ECCS Subsystems - T_{avg} more than or equal to 280°F.'' Surveillance requirement 4.5.2.f would be modified to state that opening and closing of the inspection port on the watertight enclosure for the decay heat valve pit would not require this surveillance procedure to be performed. The applicable TS bases would also be changed. Date of publication of individual notice in **Federal Register:** February 26, 1997 (62 FR 8783) Expiration date of individual notice: March 28, 1997

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

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.

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 application for amendments: August 1, 1996

Brief description of amendments: The amendments modify the Technical Specifications requirements to allow use of blind flanges during Modes 1-4 in the Calvert Cliffs 1 and 2 Containment Purge system instead of the two outboard 48-inch isolation valves. Date of issuance: March 7, 1997

Effective date: As of the date of issuance to be implemented by the end of the 1998 refueling outage for Unit 1; by the end of the 1997 refueling outage for Unit 2.

Amendment Nos.: 221 and 197

Facility Operating License Nos. DPR-53 and DPR-69: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** September 11, 1996 (61 FR 47975) The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated March 7, 1997 No significant hazards consideration comments received: No

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

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 application for amendment: December 30, 1996

Brief description of amendment: The amendment revises chemistry data for TS Figures 3.4-2 and 3.4-3 and the associated Bases.

Date of issuance: March 7, 1997

Effective date: March 7, 1997

Amendment No.: 68

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

Date of initial notice in **Federal Register:** January 29, 1997 (62 FR 4342) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 7, 1997. No significant hazards consideration comments received: No

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: September 20, 1996, as supplemented January 21, 1997.

Brief description of amendments: The amendments would update the pressure- temperature cures contained in the Dresden and Quad Cities Technical Specifications to 22 Effective Full Power Years. Date of issuance: February 28, 1997 Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 153, 148, 172 and 168

Facility Operating License Nos. DPR-19, DPR-25, DPR-29 and DPR-30. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 18, 1996 (61 FR 66703). The January 21, 1997, submittal provided additional clarifying information that did not change the original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 28, 1997 No significant hazards consideration comments received: No

Local Public Document Room location: for Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: December 6, 1996

Brief description of amendments: The amendments would change the Technical Specification (TS) by allowing a single control rod to be moved when the plant is in the Hot Shutdown or Cold Shutdown condition provided that the one-rod-out interlock is Operable and the reactor mode switch is in the refuel position.

Date of issuance: March 4, 1997

Effective date: Immediately, to be implemented within 60 days.

Amendment Nos.: 154, 149, 173, 169

Facility Operating License Nos. DPR-19, DPR-25, DPR-29 and DPR-30. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 15, 1997 (62 FR 2187). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 4, 1997. No significant hazards consideration comments received: No

Local Public Document Room location: For Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: January 6, 1997

Brief description of amendments: The amendments would change the technical specifications to clarify and maintain consistency between the operability requirements for protective instrumentation and associated automatic bypass features.

Date of issuance: March 14, 1997

Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 155, 150, 174, 170

Facility Operating License Nos. DPR-19, DPR-25, DPR-29 and DPR-30. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 12, 1997 (62 FR 6573). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 14, 1997. No significant hazards consideration comments received: No

Local Public Document Room location: For Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

Entergy Operations, Inc., Docket No. 50-313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: November 26, 1996 as supplemented by letters dated December 17, 1996, March 4, 1997, and March 10, 1997

Brief description of amendment: The amendment changes reactor coolant systems pressure/temperature limits to incorporate updated parameters and requirements.

Date of issuance: March 14, 1997

Effective date: March 14, 1997

Amendment No.: 188

Facility Operating License No. DPR-51. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 29, 1997 (62 FR 4346) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 14, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 11, 1996 as supplemented by letters dated June 18, and September 5, 1996.

Brief description of amendment: The amendment adds low-temperature overpressure protection requirements to the Technical Specifications as proposed by Generic Letter 90-06.

Date of issuance: March 7, 1997

Effective date: March 7, 1997, to be implemented within 30 days.

Amendment No.: 180

Facility Operating License No. NPF-6. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 8, 1996 (61 FR 20846) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 7, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 23, 1996, as supplemented January 8, 1997 (TSCR 245)

Brief description of amendment: The amendment updates the pressure-temperature limits up to 22, 27, and 32 effective full power years.

Date of Issuance: March 6, 1997

Effective date: March 6, 1997, to be implemented within 30 days of issuance

Amendment No.: 188

Facility Operating License No. DPR-16. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** September 11, 1996 (61 FR 47977). The January 8, 1997, letter provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated March 6, 1997 No significant hazards consideration comments received: No.

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

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

Date of application for amendment: October 17, 1996, as supplemented and modified on December 13, 1996

Brief description of amendment: The amendment revises the Operating License to reflect the transfer of Soyland Power Cooperative's 13.21-percent minority ownership of Clinton Power Station to Illinois Power Company. The Operating License has been revised to delete Soyland Power Cooperative as an owner.

Date of issuance: March 13, 1997

Effective date: March 13, 1997

Amendment No.: 114

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

Date of initial notice in **Federal Register:** November 19, 1996 (61 FR 58897) and January 29, 1997 (62 FR 4337) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 13, 1997. No significant hazards consideration comments received: No

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

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

Date of application for amendment: June 19, 1996, and supplemented September 19, 1996, and December 20, 1996.

Brief description of amendment: The amendment revises the TS to allow a permanent extension of the interim steam generator tube voltage-based repair criteria for steam generator tubes used in Cycles 13, 14 and 15 at the Donald C. Cook Nuclear Power Plant, Unit 1.

Date of issuance: March 13, 1997

Effective date: March 13, 1997, with full implementation within 45 days

Amendment No.: 215

Facility Operating License No. DPR-58. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 31, 1996 (61 FR 40022) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 13, 1997. No significant hazards consideration comments received: No. The September 19, 1996, and December 20, 1996, letters provided additional information within the scope of the original application and did not change the initial proposed no significant hazards consideration determination.

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

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan

Date of application for amendments: May 26, 1995, and supplemented September 26, 1995, August 2, 1996 and February 6, 1997

Brief description of amendments: The amendments revise the TS to allow operation of Cook Unit 1 at steam generator tube plugging levels up to 30%. Additional changes to increase operating margins for both Unit 1 and Unit 2 are also included.

Date of issuance: March 13, 1997

Effective date: March 13, 1997, with full implementation within 45 days

Amendment Nos.: 214 and 199

Facility Operating License Nos. DPR-58 and DPR-74. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 19, 1995 (60 FR 37095) The September 26, 1995, August 2, 1996, and February 6, 1997, supplements provided clarifying information that did not expand the scope of the initial application or change the staff's proposed no significant hazards determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 13, 1997. No significant hazards consideration comments received: No.

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

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

Date of amendment request: October 17, 1996

Description of amendment request: The amendment revises the Appendix A Technical Specifications relating to in-core detector system, seismic instrumentation, meteorological instrumentation, and turbine overspeed protection. The amendment deletes Limiting Conditions for Operation and Surveillance Requirements related to these instruments. The deleted requirements are to be incorporated into the Seabrook Station Technical Requirements Manual (SSTR). The associated Bases Sections are also deleted. In addition, Technical Specification 5.5 is deleted but will not be relocated to the SSTR. The amendment also redesignates Paragraph 2.J of the Seabrook Operating License as Paragraph 3, and has added new Paragraph 2.J to document the North Atlantic commitment to relocate the above mentioned Technical Specification requirements to the SSTR.

Date of issuance: March 12, 1997

Effective date: March 12, 1997

Amendment No.: 50

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

Date of initial notice in **Federal Register:** December 18, 1996 (61 FR 66713). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 12, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

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

Date of application for amendment: May 23, 1996, as supplemented July 17 and December 4, 1996

Brief description of amendment: The amendment modifies the description of the time constants associated with the Overtemperature Delta-T and Overpower Delta-T calculations used to establish the trip setpoints and the time constant used in the rate-lag controller for Steam Line Isolation, Steam Line Pressure Negative Rate-High.

Date of issuance: March 11, 1997

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

Amendment No.: 134

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

Date of initial notice in **Federal Register:** June 17, 1996 (61 FR 30639) The July 17 and December 4, 1996, letters provided additional, clarifying information that did not change the scope of the May 23, 1996, application and the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 11, 1997. 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, Connecticut 06360, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut 06385

Southern California Edison Company, et al., Docket No. 50-362, San Onofre Nuclear Generating Station, Unit No. 3, San Diego County, California

Date of application for amendment: February 7, 1997

Brief description of amendment: The amendment defers implementation of Surveillance Requirement 3.1.5.4 of Technical Specification 3.1.5, ''Control Element Assembly (CEA) Alignment,'' until the next SONGS Unit 3 shutdown, which will be no later than the upcoming Cycle 9 refueling outage (currently scheduled for April 12, 1997).

Date of issuance: March 5, 1997

Effective date: March 5, 1997

Amendment No.: 126

Facility Operating License No. NPF-15: The amendments revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: Yes (62 FR 7477 dated February 19, 1997). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by March 21, 1997, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 5, 1997.

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770

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

Dated at Rockville, Maryland, this 19th day of March 1997.

For the Nuclear Regulatory Commission

Elinor G. Adensam,

Acting Director, Division of Reactor Projects III/IV, Office of Nuclear Reactor Regulation

[Doc. 97–7508 Filed 3–25–97; 8:45 am]

BILLING CODE 7590–01–F

[Docket No. 50–409]

Lacrosse Boiling Water Reactor; Intent To Relocate Local Public Document Room

Notice is hereby by given that the Nuclear Regulatory Commission (NRC) will be relocating the local public document room (LPDR) for records pertaining to Dairyland Power Cooperative's LaCrosse Boiling Water Reactor located in Genoa, Wisconsin. The LaCrosse LPDR is currently located at the LaCrosse Public Library, 800 Main Street, LaCrosse, Wisconsin. The document collection is available in microfiche form, with paper copy indices. Library staff informed the NRC that they are no longer able to maintain the document collection and request that it be moved. This notice invites public comment on possible LPDR locations in the Genoa, Wisconsin, area.

Among the factors the NRC will consider in selecting a new location for the LPDR are the following:

(1) Whether the institution is an established document repository located near the nuclear facility with a history of impartially serving the public;

(2) The physical facilities available, including shelf space, storage space, patron workspace, copying equipment and computer access;