prepare an environmental impact statement for the proposed amendment.

For further details with respect to this proposed action, see the licensee's letter dated October 15, 1993, as supplemented by letters dated April 15, and November 10, 1994, which are available for public inspection at the Commissin's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, Mississippi 39120.

Dated at Rockville, Maryland this 9th day of February 1995.

For the Nuclear Regulatory Commission.

James R. Hall,

Senior Project Manager, Project Directorate IV–I, Division of Reactor Projects III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 95–3772 Filed 2–14–95; 8:45 am]

BILLING CODE 7590–01–M

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

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 January 20, 1995, through February 3, 1995. The last biweekly notice was published on February 1, 1995 (60 FR 6296).

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

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

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

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

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 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 March 17, 1995, 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, 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, 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 No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: January 19, 1995.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Surveillance Requirement 4.0.3 and its associated bases to provide for a delay period of up to 24 hours in which to perform a surveillance which has been discovered not to have been performed within its specified frequency. This change would adopt the requirements of NUREG–1431, ''Standard Technical Specifications, Westinghouse Plants.''

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 change will reduce the requirement to unnecessarily manipulate and challenge plant systems and equipment. The most probable result of performing a surveillance during the delay period will be to verify its conformance with Technical Specification requirements. Since this change does not affect plant design, operation, or the manner in which testing is performed, the consequences of accident scenarios postulated in the Final Safety Analysis Report will not increase. 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.

The proposed change does not introduce any new equipment, nor does it require existing systems to perform a different type of function than they are currently designed to perform. 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.

The margin of safety is neither described or prescribed for this specification. The proposed change simply provides additional time to perform a surveillance and verify that the operability of equipment is in conformance with the Technical Specification requirements. 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: R. E. Jones, General Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Project Director: William H. Bateman.

Commonwealth Edison Company, Docket Nos. 50–295 and 50–304, Zion Nuclear Power Station, Units 1 and 2, Lake County, Illinois

Date of amendment request: December 23, 1994.

Description of amendment request: The proposed amendments would increase the allowable enrichment of new fuel stored in the new fuel storage vault (NFSV), revise the enrichment description of fuel in the reactor core, and include references to documents previously approved by the staff in the

Technical Specifications that provide analytical methods used to determine core operating limits.

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:

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

The Updated Final Safety Analysis Report (UFSAR) does not consider any accidents involving the NFSV. The Fuel Handling Accidents that are analyzed (Section 15.7.4) include dropping of a spent fuel assembly onto the spent fuel pool floor and breaking of all fuel rods, and dropping of a fuel assembly inside containment onto the top of the core.

The proposed change to increase the NFSV fuel enrichment limit from 4.0 to 4.65 weight percent U–235 does not affect any of the initiators or precursors of any accident previously evaluated. The proposed change will not increase the likelihood that a transient initiating event will occur because transients are initiated by equipment malfunction and/or catastrophic system failure. Since the proposed change does not involve the introduction of new or redesigned plant equipment, failure mechanisms are not affected. As a result, the probability of occurrence of accidents previously evaluated is not significantly increased.

A new criticality analysis for the proposed change to increase the NFSV fuel enrichment limit from 4.0 to 4.65 weight percent U–235 was performed for the NFSV. It was determined that even in worst case conditions the acceptance criteria was met since the maximum K_{eff} was determined to be well below the 0.95 limit with a 95/95 probability/confidence level. The consequences of any accident, including a fuel handling accident involving the NFSV, are not significantly increased.

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

The proposed change to the Technical Specifications does not involve the addition of any new or different types of safety related equipment, nor does it involve the operation of equipment required for safe operation of the facility in a manner different from those addressed in the safety analysis. No safety related equipment or function will be altered as a result of the proposed changes. Also, the procedures governing normal plant operation and recovery from an accident are not changed by the proposed Technical Specification changes. Since no new failure modes or mechanisms are added by the proposed changes, the possibility of a new or different kind of accident is not created.

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

Plant safety margins are established through LCOs, limiting safety system settings, and safety limits specified in the Technical Specifications. There will be no changes to either the physical design of the plant or to any of these settings and limits as a result of increasing the NFSV fuel enrichment limit. The change does not involve a significant increase in the probability of occurrence or consequences of any accident previously evaluated or create the possibility of a new or different kind of accident from any previously analyzed. Additionally, the revised criticality analysis demonstrates that the maximum K_{eff} under all postulated conditions remains below the acceptance value of 0.95. Therefore, the change will not result in a significant reduction in a margin of safety.

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

The proposed change to increase the reactor core fuel enrichment range discussed in the Design Features section of Technical Specifications from ''between 2.2 to 4.0'' to ''up to 4.65'' weight percent U–235 is administrative in nature and does not affect any of the initiators or precursors of any accident previously evaluated. The proposed change will not increase the likelihood that a transient initiating event will occur because transients are initiated by equipment malfunction and/or catastrophic system failure. Since the proposed change does not involve the introduction of new or redesigned plant equipment, failure mechanisms are not affected. As a result, the probability of occurrence of accidents previously evaluated is not significantly increased.

The fuel enrichment limit of each core is determined by the core specific design and is determined to be acceptable with respect to the accident analysis by the reload analysis and is not impacted by the value specified in the description in the Design Features section of Technical Specifications. This value is only provided as the highest expected core fuel enrichment in the Design Features section discussion of the reactor core. This change is administrative in nature and does not affect the consequences of any accident previously evaluated.

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

The proposed change in the reactor core fuel enrichment description contained in the Design Features section of Technical Specifications does not involve the addition of any new or different types of safety related equipment, nor does it involve the operation of equipment required for safe operation of the facility in a manner different from those addressed in the safety analysis. No safety related equipment or function will be altered as a result of the proposed change. Also, the procedures governing normal plant operation and recovery from an accident are not changed by the proposed Technical Specification change. Since no new failure modes or mechanisms are added by the proposed change, the possibility of a new or different kind of accident is not created.

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

Plant safety margins are established through LCOs, limiting safety system settings, and safety limits specified in the Technical Specifications. There will be no changes to either the physical design of the plant or to any of these settings and limits as a result of increasing reactor core fuel enrichment value given in the Design Features section of Technical Specifications. The change does not involve a significant increase in the probability of occurrence or consequences of any accident previously evaluated or create the possibility of a new or different kind of accident from any previously analyzed.

Based on the above discussion, the ability to safely shutdown the operating unit and mitigate the consequences of all accidents previously evaluated will be maintained. Therefore, the margin of safety is not significantly affected.

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

The proposed change to add three documents to the list of documents that provide the analytical methods to determine core operating limits is administrative in nature and does not affect any of the initiators or precursors of any accident previously evaluated. The proposed change will not increase the likelihood that a transient initiating event will occur because transients are initiated by equipment malfunction and/or catastrophic system failure. Since the proposed change does not involve the introduction of new or redesigned plant equipment, failure mechanisms are not affected.

The documents have been previously reviewed and approved by the NRC and it was determined that they provide an acceptable means to determine core operating limits. As a result, the probability of occurrence of accidents previously evaluated is not significantly increased. Since the documents provide NRC approved methodologies for determining core operating limits, the addition of the documents to Technical Specifications or use of the documents to determine core operating limits will not significantly increase the consequences of any accident previously evaluated.

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

The proposed change to add three documents to the list of documents that provide the analytical methods to determine core operating limits is administrative in nature and does not involve the addition of any new or different types of safety related equipment, nor does it involve the operation of equipment required for safe operation of the facility in a manner different from those addressed in the safety analysis. No safety related equipment or function will be altered as a result of the proposed changes. Also, the procedures governing normal plant operation and recovery from an accident are not changed by the proposed Technical Specification changes. Since no new failure modes or mechanisms are added by the proposed changes, the possibility of a new or different kind of accident is not created.

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

Plant safety margins are established through LCOs, limiting safety system settings, and safety limits specified in the Technical Specifications. There will be no changes to either the physical design of the plant or to any of these settings and limits as a result of adding references to the new documents. The ability to mitigate the consequences of all accidents previously evaluated will be maintained. Therefore, the margin of safety is not significantly affected.

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: Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60690.

NRC Project Director: Robert A. Capra.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York Date of amendment request: September 19, 1994.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 4.4.A.3 to reference the testing frequency requirements of 10 CFR Part 50, Appendix J, and to state that NRC approved exemptions to the applicable regulatory requirements are permitted. This proposed administrative revision simply deletes the paraphrased language and directly references Appendix J.

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

Criterion 1—Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed change will provide a one-time exemption from the 10 CFR [Part] 50, Appendix J Section III.D.1.(a) leak rate test schedule requirement. This change will allow for a one-time test interval for Type A Integrated Leak Rate Tests (ILRTs) of approximately 70 months.

Leak rate testing is not an initiating event in any accident, therefore this proposed change does not involve a significant increase in the probability of a previously evaluated accident.

Type A tests are capable of detecting both local leak paths and gross containment failure paths. The history at IP–2 [Indian Point 2] demonstrates that Type B and C Local Leak Rate Tests (LLRTs) have consistently detected any excessive local leakages.

Administrative controls govern the maintenance and testing of containment penetrations such that the probability of excessive penetration leakage due to improper maintenance or valve misalignment is very low. Following maintenance on any containment penetration, an LLRT is performed to ensure acceptable leakage levels, following any LLRT on a containment isolation valve, an independent valve alignment check is performed. Therefore, Type A testing is not necessary to ensure acceptable leakage rates through containment penetrations.

While Type A testing is not necessary to ensure acceptable leakage rates through containment penetrations, Type A testing is necessary to demonstrate that there are no gross containment failures. Structural failure of the containment is considered to be a very unlikely event, and in fact, since IP–2 has been in operation it has never failed a Type A ILRT. Therefore, a one-time exemption increasing the interval for performing an ILRT should not result in a significant decrease in the confidence in the leak tightness of the containment structure.

The proposed change also revises Technical Specification 4.4.A.3 to reference the testing frequency requirements of 10 CFR [Part] 50, Appendix J, and to state that NRC approved exemptions to the applicable regulatory requirements are permitted. The current language of TS 4.4.A.3 paraphrases the requirements of Section III.D.1.(a) of Appendix J. The proposed administrative revision simply deletes the paraphrased language and directly references Appendix J. No new requirements are added, nor are any existing requirements deleted. Any specific changes to the requirements of Section III.D.1.(a) will require a submittal from Consolidated Edison under 10 CFR 50.12 and subsequent review and approval by the NRC prior to implementation. The proposed change is stated generically to avoid the need for further TS changes if different exemptions are approved in the future.

The proposed change, in itself, does not affect reactor operations or accident analysis and has no radiological consequences. The change provides clarification so that future Technical Specifications changes will not be necessary to correspond to applicable NRC approved exemptions from the requirements of Appendix J.

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

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

The proposed exemption request does not affect normal plant operations or configuration, nor does it affect leak rate test methods. The proposed change allows a one-time test interval of approximately 70 months for the ILRT. Given the test history of IP–2 of no Type A test failures during plant lifetime, the relaxation in schedule should not significantly decrease the confidence in the leak tightness of the containment.

The proposed Technical Specification amendment provides clarification to a specification that paraphrases a codified requirement.

Since the proposed change would not change the design, configuration or method of operation of the plant, it would not create the possibility of a new or different kind of accident from any previously evaluated.

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

The purpose of the existing schedule for ILRTs is to ensure that the release of radioactive materials will be restricted to those leak paths and leak rates assumed in accident analyses. The relaxed schedule for ILRTs does not allow for relaxation of Type B and C LLRTs. Therefore, methods for detecting local containment leak paths and leak rates are unaffected by this proposed change. Given that the test history for ILRTs shows no failure during plant life, a one-time increase of the test interval does not lead to a significant probability of creating a new leakage path or increased leakage rates, and the margin of safety inherent in existing accident analyses is maintained.

The proposed Technical Specification change is administrative and clarifies the relationship between the requirements of TS 4.4.A.3, Appendix J and any approved exemptions to Appendix J. It does not, in itself, change a safety limit, an LCO [limiting condition for operation], or a surveillance requirement on equipment required to operate the plant. The NRC will directly approve any proposed change or exemption to [Section] III.D.1.(a) of Appendix J prior to implementation.

Therefore, this change does not involve a significant reduction in the margin of safety.

Based on the Safety Analysis, it is concluded that: (1) The proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92 and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change. Moreover, because this action does not involve a significant hazards consideration, it will also not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

Although the licensee has included an evaluation of a proposed exemption to 10 CFR part 50, Appendix J requirements in the above determination of no significant hazards consideration, only the part related to the amendment is pertinent to this notice of proposed amendment. The exemption request will be considered as a separate matter on its own merits. The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: Ledyard B. Marsh

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

Date of amendment request: October 31, 1994

Description of amendment request: The requested amendments would remove the stroke times for the steam generator power operated relief valves (PORVs) from Technical Specification (TS) Tables 3.6–2a and 3.6–2b. The PORVs are part of the main steam vent to atmosphere system. The PORV actuators have difficulty developing enough closing thrust to adequately overcome all of the friction loads within the valves; therefore, difficulty exists in consistently meeting the present 5-second closing stroke time requirement. The licensee requests the proposed change on the basis that the PORVs do not receive an actual containment isolation signal; therefore, it is justified to remove the stroke times from TS Tables 3.6–2a and 3.6–2b.

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 48 FR 14870, the Commission has set forth examples of amendments that are considered not likely to involve significant hazards considerations. Example (vi) describes a change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan. In this case, the proposed amendment is similar to example (vi) in that it removes the required isolation time of the steam generator PORVs from TS Tables 3.6–2a and 3.6–2b; however, no adverse impact upon accident analyses is created as a result.

Criterion 1

The requested amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated. The effects of the delays in isolation times on the various transients affected have been analyzed and found to be acceptable. Since these valves do not receive a containment isolation signal, and no credit is taken for operation of these valves in the dose analysis for a containment isolation function, a maximum stroke time does not apply for containment isolation.

Criterion 2

The requested amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. SV PORV closure (provided the valves are not already closed at the start of the transient) is a response to a transient already in progress. The possibility of a spurious SV PORV opening will not be affected by the requested amendments. No equipment or component reconfiguration will occur as a result of this change. Finally, no changes to plant procedures are being made which would affect any accident causal mechanisms.

Criterion 3

The requested amendments will not involve a significant reduction in a margin of safety. The isolation times which are applicable to these valves are specified in TS Table 3.3–5, Engineered Safety Features Response Times. The effects of the isolation of these valves were evaluated based on their ESF function, not a containment isolation function, and determined to be acceptable.

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

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

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

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: June 13, 1994, as supplemented August 15, 1994.

Description of amendment request: The proposed changes would increase the initial fuel enrichment limit from a current maximum of 4.0 weight % to 4.75 weight % and establish new loading patterns for new and irradiated fuel in the spent fuel pool to accommodate this increase. These changes would also increase the efficiency of fuel storage cell use in the spent fuel pools and provide additional flexibility to the reload design efforts at Duke Power Company, while at the same time maintaining sufficient criticality safety margin and decay heat removal capabilities.

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.

There is no increase in the probability or consequences of an accident in the new fuel vault since the only credible accidents for this area are criticality accidents and it has been shown that calculated, worst case K_{eff} for this area is $\leq$0.95 under all conditions.

There is no increase in the probability of a fuel drop accident in the Spent Fuel Storage Pool since the mass of an assembly will not be affected by the increase in fuel enrichment. The likelihood of other accidents, previously evaluated and described in Section 9.1.2 of the FSAR [Final Safety Analysis Report], is also not affected by the proposed changes. In fact, it could be postulated that since the increase in fuel enrichment will allow for extended fuel cycles, there will be a decrease in fuel movement and the probability of an accident may likewise be decreased. There is also no increase in the consequences of a fuel drop accident in the Spent Fuel Pool since the fission product inventory of individual fuel assemblies will not change significantly as a result of increased initial enrichment. In addition, no change to safety related systems is being made. Therefore, the consequences of a fuel rupture accident remain unchanged. Also, it has been shown that k_{eff} is $\leq$0.95, under all conditions therefore, the consequences of a criticality accident remain unchanged as well.

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

The proposed changes do not create the possibility of a new or different kind of accident since fuel handling accidents (fuel drop and misplacement) are not new or different kinds of accidents. Fuel handling accidents are already discussed in the FSAR for fuel with enrichments up to 4.1 weight %. As described in Section VI.9 of Attachment IV, additional analyses have been performed for fuel with enrichment up to 4.75 weight %. Worst case misloading accidents associated with the new loading patterns were evaluated. For all possible misloading accidents the negative reactivity provided by soluble boron maintains k_{eff} $\leq$0.95. of safety.

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

The proposed change does not involve a significant reduction in the margin of safety since, in all cases, a k_{eff} $\leq$0.95 is being maintained. Criticality analyses have been performed which show that the new fuel storage vault will remain subcritical under a variety of moderation conditions, from fully flooded to optimum moderation. As discussed above, the Spent Fuel Pool will remain sufficiently subcritical during any fuel misplacement accident.

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

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

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

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: November 11, 1994, as supplemented January 30, 1995.

Description of amendment request: The amendments would revise the Technical Specifications Design Features section to establish restricted loading patterns and associated burnup criteria for placing fuel in the Oconee Spent Fuel Pools. These changes are necessary to address two new fuel designs which have increased initial fuel enrichment and therefore cannot be stored in the spent fuel pools under existing 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:

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

Each accident analysis addressed in the Oconee Final Safety Analysis Report (FSAR) has been examined with respect to changes in Cycle 15 parameters to determine the effect of the Cycle 16 reload and to ensure that the acceptance criteria of the FSAR safety analyses remain satisfied. The transient evaluation of Cycle 16 is considered to be bounded by previously accepted analyses. Section 7 of the Reload Report addresses ''Accident and Transient Analysis'' for this core reload.

There is no increase in the probability or consequences of an accident due to the spent fuel storage restrictions proposed in this amendment request. It has been shown that the calculated, worst case k_{eff} for this area is [less than or equal to] 0.95 under all conditions. There is no increase in the probability of a fuel drop accident in the SFP [spent fuel pool] since the mass of the new assemblies is not significantly different from the mass of the old assemblies. The likelihood of other accidents, previously evaluated and described in the FSAR, is also not affected by the proposed changes. In fact, it could be postulated that since the increase in fuel enrichment will allow for extended fuel cycle lengths, there will be a decrease in fuel movement and the probability of an accident may actually be reduced. There is also no increase in the consequences of a fuel rod drop accident in the SFP since the fission product inventory of individual fuel assemblies will not change significantly as a result of increasing the initial enrichment. In addition, no change to safety related systems is being made. Therefore, the consequences of a fuel rupture accident remain unchanged. In addition, it has been shown that k_{eff} is [less than or equal to] 0.95 under all conditions. Therefore, the consequences of a criticality accident in the SFP remain unchanged as well. The above analysis ensures that the proposed reload amendment request will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The analyses performed in support of this reload are in accordance with the NRC approved methods delineated in Specification 6.9.2. The predicted operating characteristics of Oconee 3 Cycle 16 are similar to previously licensed designs. The Mark B10T and Mark B11 fuel assembly designs remain mechanically compatible with all fuel handling equipment. Therefore, no new or different kind of fuel handling accident is created by the proposed amendment request.

Section 15.11 of the Oconee FSAR states that the refueling boron concentration is maintained such that a criticality accident during refueling is not considered credible. The proposed amendment request continues to assure that a criticality accident in the SFP or during refueling is not credible. The double contingency principle discussed in ANSI N–16.1–1975 and the April 1978 NRC letter allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. Thus, by requiring a minimum boron concentration in the SFP, a criticality accident caused by violating the SFP storage restrictions is not considered credible. Therefore, the proposed amendment request does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The Oconee 3 Cycle 16 design was performed using the NRC approved methods given in Specification 6.9.2. The safety limits for Oconee 3 Cycle 16 are unchanged from previous cycles. The limits and margins summarized in the Oconee 3 Cycle 16 Reload Report are well within the allowable limits and requirements, and reflect no reductions to any margins of safety.

The proposed change does not involve a significant reduction in the margin of safety related to SFP criticality. In all cases, a k_{eff} [less than or equal to] 0.95 is maintained. Criticality analyses have been performed which show that the SFP will remain sufficiently subcritical during any fuel misplacement accident. In summary 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 amendment request involves no significant hazards consideration.

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

Attorney for licensee: J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20036.

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: January 6, 1995

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) 3/4.8.1.1, ''AC Sources-Operating,'' and 3/4.8.1.2, ''AC Sources-Shutdown,'' to (1) revise the minimum quantity of fuel oil required in the day tanks and the storage tanks, (2) add specific actions to be taken if the storage tank levels fall below minimum requirements, (3) revise and relocate to the associated Bases the fuel oil sampling and testing criteria, and (4) add specific actions to be taken if the fuel oil properties do not meet specified limits. The proposed amendment would also revise TS 6.8.4, ''Programs,'' to add a requirement for a diesel fuel oil testing program. The licensee stated that the proposed changes are consistent with the NRC's Improved Standard Technical Specifications (NUREG–1434).

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

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

The diesel generators are not initiators or precursors to an accident previously evaluated. The diesel generators are required to provide onsite power to safe shutdown loads as assumed in the accident analysis. Therefore, the proposed changes to the diesel generator fuel oil specifications cannot significantly affect the probability of a previously evaluated accident.

The proposed change to the minimum required diesel generator fuel oil levels is based on updated calculations of fuel consumption rates. Because the updated calculations assume a lower consumption rate, the new minimum fuel oil levels are lower but still assure that a seven-day fuel oil capacity is available. Accordingly, the proposed change has no effect on the operation of the diesel generator. The proposed change to allow 48-hours to restore diesel generator fuel oil to the minimum required level does not affect short-term diesel generator operability and is acceptable based on the remaining fuel oil capacity (>6 days), initiating the process for procuring additional fuel and the low probability of an event requiring a diesel generator during this interval. Also, the proposed allowance of a limited time to restore diesel fuel oil properties to required limits will not affect the short-term operability of the diesel generator. Even with minor degradation of the fuel oil properties, the diesels will start and perform their intended function. Relocation of the testing requirements to the bases and adding a description of the Diesel Fuel Oil Testing Program to the Administrative Control section are administrative changes. The diesel fuel oil will continue to be sampled and tested in a manner to assure its quality. In summary, the changes will not adversely affect the performance or the ability of the diesel generators to perform their intended function. Therefore, the proposed changes will not significantly increase the consequences of an accident previously evaluated.

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

The proposed changes will revise the minimum required diesel generator fuel oil levels and requirements associated with diesel generator fuel oil properties.

The changes do not introduce any new accident precursors and do not involve any alterations to plant configurations which could initiate a new or different kind of accident. The proposed changes do not affect the short-term operability of the diesel generator. In addition, the operability of the diesel generators is assured by periodic testing and preventive maintenance. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Safety margins are established through safety analyses. These analyses assume that at least one diesel generator will start and load whenever offsite power is lost. The proposed change to the minimum required diesel generator fuel oil levels is based on updated calculations of fuel consumption rates. The updated calculations use the guidance delineated in Regulatory Guide 1.137 which is based on time-dependent loads of the diesel-generators during design basis events. Calculations based on time dependent loads result in new minimum fuel oil levels which are lower. This change has no effect on the operation of the diesel generator or on a margin of safety. The allowance of a limited time to restore the fuel oil levels, or to analyze and restore fuel oil properties to required limits, is justified since the short term operability of the diesel generators is not affected. Relocation of the fuel oil testing requirements to the Bases does not affect the quality of the fuel oil. The 10CFR50.59 process will assure that future changes to the Bases will maintain the current margins of safety, and that the diesel fuel oil will continue to be sampled and tested in such a manner as to assure its quality. Adding a description of the Diesel Fuel Oil Testing Program to the Administrative Control section of Technical Specifications are administrative. Therefore, the diesel generator will continue to operate as analyzed and there will not be a significant reduction in a margin of safety.

The proposed changes are further justified in that they are consistent with the requirements of the Improved Standard Technical Specifications (NUREG–1434).

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

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

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

NRC Project Director: Ledyard B. Marsh.

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

Date of amendment request: January 6, 1995.

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) 3/4.3.7.5, ''Accident Monitoring Instrumentation,'' and TS 3/4.4.2, ''Safety/Relief Valves.'' TS 3/4.3.7.5 would be revised to delete certain instruments not classified as Category 1 (Type A or non-Type A) as defined in Regulatory Guide 1.97 and to delete the requirement that accident monitoring instrumentation be operable in Operational Condition 3. The ACTIONS of TS Table 3.3.7.5–1 would be revised to allow 30 days to restore one inoperable channel and 7 days to restore two inoperable channels. TS 3.3.7.5 would be revised to add an exception to the requirements of TS 3.0.4. In addition, editorial changes would be made to TS Tables 3.3.7.5–1 and 4.3.7.5–1 for consistency and clarity.

The proposed amendment would also revise TS 3/4.4.2 to remove requirements related to safety/relief valve acoustic monitors to be consistent with the proposed changes to TS Tables 3.3.7.5–1 and 4.3.7.5–1.

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:

The operation of NMP2 [Nine Mile Point Nuclear Station Unit 2] in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

PAM [Post-Accident Monitoring] instruments are used to help guide operator response to postulated accidents. Thus, the status or operability of PAM instrumentation does not affect the probability of previously analyzed accidents. The non-Category 1 PAM instruments being removed from the Technical Specifications do not meet any of the Commission's screening criteria and are not of controlling importance to safety or necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to public health and safety. The operability of critical parameters necessary to assure proper response to previously analyzed accidents (i.e., Category 1 instruments) is still controlled by the Technical Specifications. Thus, deleting non-Category 1 instruments will not increase the consequences of any accident previously evaluated.

PAM instruments are related to the diagnosis and preplanned actions required to mitigate DBAs [Design Basis Accidents] assumed to occur in Operational Conditions 1 and 2. A DBA during Operational Condition 3 is extremely unlikely. The requirement to maintain the Reactor Water Level, Suppression Pool Water Level and Drywell High Range Radiation Monitor instrumentation operable in Operational Condition 3 will be deleted. Because Suppression Pool Water Level indication will no longer be required in Operational Condition 3, its ACTION requirement was revised to delete the requirement to place the plant in COLD SHUTDOWN, Operational Condition 4. This is consistent with ITS [Improved Standard Technical Specifications] which requires that the plant be brought to an operational condition in which the LCO [Limiting Condition for Operation] does not apply if a required action cannot be met. Therefore, deleting the requirement that PAM instruments be operable during Operational Condition 3 and changing the ACTION requirement for Suppression Pool Water Level Monitoring does not affect the probability or consequences of an accident.

The passive nature of the Category 1 PAM instruments (i.e., those instruments that initiate no critical automatic action) and the alternate means available to obtain the required information assure an acceptable level of safety is maintained during operation with instrument channels out of service. Since an acceptable level of safety is maintained with inoperable channels, plant startup or operation with inoperable channels will not alter plant response to analyzed accidents. Thus, the proposed changes to the required ACTIONS and the proposed exemption to Specification 3.0.4 will not increase the consequences of analyzed events.

The proposed changes to the requirements for PCIV [Primary

Containment Isolation Valve] indication are consistent with the proposed required ACTIONS. Position indication will still be required for each operable PCIV and penetrations without adequate PCIV indication status will be isolated, thus assuring containment integrity in the event of an accident. Deletion of the ''Minimum Required Actions'' column in Table 3.3.7.5–1 is consistent with the proposed ACTIONS for LCO 3.3.7.5, since compensatory actions are based on compliance with the ''Required Number of Channels.'' Deleting the ''Applicable Operating Conditions'' column is consistent with the proposed changes and other NMP2 Technical Specifications sections. Finally, referencing Specification 4.0.5 is an administrative change which does not alter any existing surveillance requirements for the safety relief valves.

In aggregate, the proposed changes do not affect the plant in a way that could directly contribute to causing or mitigating the effects of an accident. Therefore, the operation of NMP2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The operation of NMP2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not represent a physical change to the plant as described in the NMP2 USAR [Updated Safety Analysis Report]. The proposed changes do not modify any plant equipment and the initial conditions used for the design basis accident analysis are still valid. Thus, no potential initiating events are created which would cause any new or different kinds of accidents. PAM instrumentation is used to guide operator response during postulated accidents. Those PAM instruments considered of controlling importance to safety are retained in the Technical Specifications. Thus, plant response to previously analyzed events is not altered so as to create any new or different kinds of accidents. Therefore, operation of Nine Mile Point Unit 2 in accordance with the proposed change will not create the possibility of a new or different kind of accident from any previously assessed.

The operation of NMP2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The non-Category 1 PAM instruments being removed from the Technical Specifications do not meet any of the Commission's screening criteria. That is, the instruments being proposed for removal are not of controlling importance to safety or necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to public health and safety. Thus, they are not critical to any margin of safety.

PAM instruments are related to the diagnosis and preplanned actions required to mitigate DBAs assumed to occur in Operational Conditions 1 and 2. A DBA during Operational Condition 3 is extremely unlikely. The requirement to maintain the Reactor Water Level, Suppression Pool Water Level and Drywell High Range Radiation Monitor instrumentation operable in Operational Condition 3 will be deleted. Because Suppression Pool Water Level indication will no longer be required in Operational Condition 3, its ACTION requirement was revised to delete the requirement to place the plant in COLD SHUTDOWN, Operational Condition 4. This is consistent with the ITS, which requires that the plant be brought to an operational condition in which the LCO does not apply if a required action cannot be met. Therefore, deleting the requirement that PAM instruments be operable during Operational Condition 3 and changing the ACTION requirement for Suppression Pool Water Level Monitoring does not significantly reduce a margin of safety.

Since the Category 1 PAM instruments are passive in nature (i.e., no critical automatic action is assumed to occur from these instruments) and alternate means exist to obtain the required information, an acceptable level of safety is assured when instrument channels are out of service. Also, the probability of an event requiring PAM instrumentation is low. Continued operation with one channel out of service, and limited plant operation with two channels out of service, does not compromise plant safety margins. An acceptable level of safety is maintained during plant startups and operation with instrument channels out of service. Thus, the proposed changes to the required ACTIONS and the proposed exemption to Specification 3.0.4 will not significantly reduce a margin of safety.

The proposed changes to PCIV indication will assure correct implementation of the ACTIONS discussed above. Isolating the flow path associated with one or two inoperable PCIV indication channels is conservative since the subject valve will be positioned as required to assure primary containment integrity. The remaining editorial changes are administrative in nature and by definition do not affect safety margins. Deleting the ''Minimum Operable Channels'' and ''Applicable Operating Conditions'' columns is consistent with the proposed changes. Finally, referencing the requirements of Specification 4.0.5 is an administrative change and by definition does not reduce the margin of safety.

Therefore, the operation of NMP2 in accordance with the proposed change will not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Ledyard B. Marsh.

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

Date of amendment request: December 9, 1994.

Description of amendment request: The proposed changes incorporate NRC recommendations contained in Generic Letter 93–05 related to the diesel generator (DG) surveillance requirements and other DG surveillance requirements related to the cold starts. The proposed changes to the DG operability testing surveillance requirements are consistent with the intent of GL 93–05 however vary in some particulars, because of circumstances specific to Millstone 3. The proposed changes will modify the requirement for the DG operability testing when the other DG is inoperable, delete the requirement for DG operability testing when one or both offsite AC sources are inoperable, eliminate fast loading of DGs except for the 18-month test, and modify the hot restart test from the 24-hour loaded test run for the DGs.

Basis for proposed no significant hazards consideration determination:

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

* * * The proposed changes do not involve a SHC because the changes would not:

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

The proposed changes revise the action requirements regarding operability testing of a non-affected DG when the other DG is inoperable, delete the requirement for operability testing of the DGs when one or both offsite AC sources are inoperable and eliminate the fast loading of DGs except for the 18-month test. These changes will improve DGs performance by reducing the number of unnecessary quick starts and by requiring more appropriate testing of the DGs when there is a potential for common mode failure. The proposed change, to revise the method of verifying DG hot restart capability after a 24-hour run without loading the DG with LOP/SI [loss of offsite power/safety injection] load, meets an intent of Regulatory Guide 1.108, Position C.2.a.5, which states the purpose of the test as to ''demonstrate functional capability at full load temperature conditions.'' Functional capability of the DG can be adequately demonstrated by manually or automatically restarting the DG within five minutes after a 24-hour test run without loading it with LOP/SI loads, provided that a full load temperature condition is maintained prior to restart. The proposed DG restart method does not reduce the effectiveness of the test. The proposed revisions of the DG surveillance requirements will not increase the probability of an accident and it will not change the response of the DG to a LOP as described in the Millstone Unit No. 3 FSAR. Since the plant response to an accident will not change, there is no change in the potential for an increase in the consequences of an accident previously analyzed.

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

The proposed changes of the DG surveillance requirements and operability testing requirements do not affect the operation or response of any plant equipment or introduce any new failure mechanisms. The proposed changes do not affect the test results and the DGs will be verified to be operable and their response to a loss of voltage will be unchanged. The plant equipment will respond per the design and analyses and there will not be a malfunction of a new or any type introduced by the revision to the DG surveillance requirements. As such, the changes do not create the possibility of a new or different kind of accident previously evaluated.

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

The bases of Technical Specification 3/4.8, ''Electrical Power Systems,'' state that the operability of the AC and DC power systems and associated distribution systems ensure that sufficient power will be available to supply the safety-related equipment required for safe shut down and mitigation and control of accident conditions. The bases also state that the surveillance requirements for determining the operability of the DGs are in accordance with the recommendations of Regulatory Guide 1.108, Revision 1. The revisions of the surveillance requirements establishes tests that will continue to verify that the DGs are operable and the testing will still meet the intent of Regulatory Guide 1.108, Revision 1. Operable DGs ensure that the assumptions in the bases of the Technical Specifications are not affected and ensure that the margin of safety is not reduced. Therefore, the assumptions in the bases of the technical specifications are not affected and these changes do not result in a significant reduction in the margin of safety.

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

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

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: Phillip F. McKee.

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

Date of amendment request: December 14, 1994.

Description of amendment request: The proposed amendment would revise the Millstone Unit No. 3 Technical Specifications by:

1. Increasing the upper bound of the overall containment integrated leakage rate required by Technical Specification 3.6.1.2.a from 0.3 wt. % per day to 0.65 wt. % per day of the containment air per 24 hours at design basis pressure.

2. Revising Technical Specification 4.6.6.1.d.3 by providing more margin with respect to the drawdown time for secondary containment vacuum.

3. Revising Bases Section 3/4.7.9 to reflect the above 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 (SHC), which is presented below:

* * * The proposed changes do not involve an SHC because the changes would not:

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

* * * There is a reasonable assurance that the modified criteria for the negative pressure in the secondary containment boundary proposed via the proposed change (i.e., a negative pressure of 0.1 inches in one minute and a negative pressure of 0.4 inches within the next two minutes), can be accomplished in the prescribed time.

Extension of the time allowed to achieve the final drawdown of secondary containment from 120 seconds to 180 seconds (these times include the diesel generator start and load time of approximately 11 seconds) will have a negligible impact on heating and cooling. Plant experience has shown that heatup and cooldown of thick-walled concrete structures, such as the Millstone Unit No. 3 auxiliary building, is a relatively slow process. Also, natural convection within the auxiliary building tends to stabilize temperatures. Following an accident signal, ventilation equipment is restarted promptly. Therefore, heatup or cooldown, during short periods while ventilation fans and/or heaters are inactive, is insignificant and can be neglected.

The proposed change to reinstate the containment integrated leakage rate at the design basis pressure from 0.3 wt % per day to 0.65 wt % per day has been evaluated to determine the impact to the Appendix J requirements for Type A, B and C Testing. In addition, the radiological consequence evaluation also addressed the increase in L_a (i.e., from 0.3 wt % per day to 0.65 wt % per day).

On October 12, 1993, Millstone Unit No. 3 successfully conducted the second

Type A test in the first 10-year service period. Test results indicated that the ''As-Found'' and ''As-Left'' ILRTs [integrated leakage rate tests] passed the technical specification acceptance criteria. The ''As-Found'' value was 0.1327 weight percent per day and the ''As-Left'' value was 0.1313 weight percent per day. These values represent 27.2% and 26.9% of the technical specification criterion of 0.4875 wt % per day (0.75 L_a), based on L_a equal to 0.65 wt % per day, respectively. In addition, as of October 9, 1993, the total Type B and C ''As-Found'' and ''As-Left'' leakage results were 0.099 wt % per day, and 0.084 wt % per day, respectively. These values represent approximately 25.3% and 21.5% of the technical specification limit of 0.39% wt % per day (0.6 L_a), based on L_a equal of 0.65 wt % per day, respectively. Correspondingly, the 1993 Type A, B, and C test results indicate that the ''As-Found'' and ''As-Left'' result in each test case was below the existing Technical Specification limit of 0.3 wt % per day. This further demonstrates the overall leakage integrity of the containment and its boundaries.

Based on the relatively low ''As-Left'' ILRT leakage rate (i.e., 0.1313 wt % per day is well below the existing technical specification limit of 0.225 wt % per day (0.75 L_a), based on L_a equal to 0.3 wt % per day), which represents the overall containment integrated leakage rate for the containment prior to start-up, there is reasonable assurance that containment integrity will be maintained below the allowable leakage rate limit of 0.65 wt % per day. In addition, the total Type B and C ''As-Left'' leakage result of 0.084 wt % per day (this is well below the existing technical specification limit of 0.18 wt % per day (0.6 L_a), based on L_a equal to 0.3 wt % per day), provides further assurance that leakage, based on individual penetration, will be maintained within sufficient margin of the leakage limits.

Because the last Type A, B, and C tests were performed under the technical specification limit of 0.65 wt % per day, the proposed change to restore L_a to 0.65 wt % per day has no impact to these systems from a leakage allowance perspective. As indicated above, the previous test results met the technical specification leakage limits (based on 0.65 wt % per day) within sufficient margin and, therefore, would not present any challenge to these leakage limits.

NNECO has evaluated the proposed changes to Surveillance Requirement 4.6.6.1.d.3 that increase the time to draw a final required negative pressure as measured at the 24'-6'' elevation of the auxiliary building in conjunction with the proposed change to reinstate the containment integrated leakage rate of 0.65 wt % per day to determine the impact on the offsite doses following a LOCA. The calculated radiological doses are, in most cases, less than the previously calculated doses (i.e., EAB [exclusion area boundary] and LPZ [low-population zone] doses) and are within the 10CFR100 limits. Previously, the EAB thyroid and whole body doses as documented in the November 4, 1993, submittal were calculated to be 141 REM and 9.4 REM respectively, while the previously docketed (i.e., the November 4, 1993, submittal) LPZ doses to the thyroid and whole body were calculated to be 29.8 REM and 1.7 REM respectively. Utilizing the revised application of containment recirculation spray DF, the EAB thyroid and whole body doses were calculated to be 61 REM and 16.7 REM, respectively, and the LPZ thyroid and whole body doses were calculated to be 10.9 REM and 2.8 REM respectively. The assumptions used in the above radiological dose calculations are provided in Attachment 1. It is noted that a LOCA at Millstone Unit No. 3 is also one of the bounding accidents for the Millstone Unit No. 3 control room, Millstone Unit No. 2 control room, and the Millstone Technical Support Center habitability analysis. Therefore, the doses for these areas were recalculated and are presented in the Safety Assessment section above. The Millstone Unit No. 1 control room and the Emergency Operating Facility doses are bounded by the Millstone Unit No. 1 LOCA calculations.

The Millstone Unit Nos. 2 and 3 control rooms and Millstone Technical Support Center doses were not recalculated in 1993 (i.e., November 4, 1993, submittal) since EAB/LPZ doses proved that the releases were less than the 1990 submittal. In summary, all control room and Technical Support Center doses are within the guidelines of GDC 19. Therefore, the proposed changes do not result in an increase in consequences of an accident (i.e., a LOCA) previously analyzed.

The proposed changes to Bases Section 3/4.6.6 do not have any safety impact since they only reflect the changes proposed to Surveillance Requirement 4.6.6.1.d.3.

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

The proposed changes do not compromise the ability of the SLCRS [supplementary leak collection and release system] and ABFS [auxiliary building filter system] to mitigate the consequences of an accident. The proposed changes do not make any physical or operational changes to existing plant structures, systems or components. The proposed changes do not introduce any new or unique operational modes or accident precursors. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

NNECO has evaluated the proposed changes to Surveillance Requirement 4.6.6.1.d.3 that increase the time to draw a final required negative pressure as measured at the 24'–6'' elevation of the auxiliary building in conjunction with the proposed change to reinstate the containment integrated leakage rate of 0.65 wt % per day to determine the impact on the offsite doses following a LOCA. The calculated radiological doses are, in most cases, less than the previously calculated doses and these doses are within the 10CFR100 limits. All control rooms and technical support center doses are within the guidelines of GDC 19. 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 amendment request involves no significant hazards consideration.

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

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: Phillip F. McKee.

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

Date of amendment request: December 23, 1994.

Description of amendment request: The proposed amendment would change the acceptance criteria for the peak transient generator voltage from 4784 volts to 5000 volts during full load rejection tests of the diesel generator (DG), and delete the 10-year surveillance requirement to perform a

110% pressure test of the DG fuel oil system.

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

* * * The proposed changes do not involve a SHC because the changes would not:

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

DG Full-Load Rejection Test

NNECO is proposing to modify Surveillance Requirement 4.8.1.1.2.g.3 of the Millstone Unit No. 3 Technical Specifications by changing the acceptable transient voltage to 5000 volts from 4784 volts. This change will permit the DG full load rejection tests to be performed at realistic plant conditions using a power factor that will envelope the calculated power factor during the worst kW loading conditions. The transient voltage of 5000 volts is within the normal design limits of the DGs.

The proposed change does not alter the intent of the surveillance, does not involve any physical changes to the plant, does not alter the way any structure, system, or component functions, and does not modify the manner in which the plant is operated. As such, the proposed change to Surveillance Requirement 4.8.1.1.2.g.3 will not degrade the capability of the DGs to perform their intended safety function, and will not reduce the availability of the DGs. Actually, the proposed change will increase the effectiveness of the full load rejection tests, because the DGs will be tested in a configuration that is closer to the design basis conditions.

Pressure Test of the DG Fuel Oil System

The DG fuel oil system is classified as an ASME Code Class 3 system in accordance with the guidance of Regulatory Guide 1.26, ''Quality Group Classification and Standards for Water-, Steam-, and Radioactive-waste Components of Nuclear Power Plants.'' Surveillance Requirement 4.0.5 requires the testing of ASME Class 1, 2, and 3 components in accordance with Section XI of the ASME Code. Surveillance Requirement 4.8.1.1.2.i.2 is redundant to the ASME Section XI pressure test requirements of Surveillance Requirement 4.0.5. Additionally, the DG fuel oil tank cannot be tested in the configuration required by Surveillance Requirement 4.8.1.1.2.i.2, because the tanks are vented to the atmosphere and the vent cannot be isolated. Therefore, NNECO is proposing to delete Surveillance Requirement 4.8.1.1.2.i.2.

The proposed change does not modify the manner in which the DGs respond to an accident. Also, the proposed change does not reduce the reliability of the DGs.

Conclusion

Based on the above, the proposed changes to Surveillance Requirements 4.8.1.1.2.g.3 and 4.8.1.1.2.i.2 of the Millstone Unit No. 3 Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

DG Full-Load Rejection Test

The DGs are required to operate in response to a loss of offsite power. Their failure cannot initiate an accident. Additionally, the proposed change to Surveillance Requirement 4.8.1.1.2.g.3 does not affect the operation or response of any plant structure, system, or component, and it does not introduce any new failure mechanisms.

Pressure Test of the DG Fuel Oil System

The proposed change to Surveillance Requirement 4.8.1.1.2.i.2 does not affect the design or function of the DG fuel oil system. Failure of the DG fuel oil system would not initiate an accident.

Conclusion

Based on the above, the proposed changes to Surveillance Requirements 4.8.1.1.2.g.3 and 4.8.1.1.2.i.2 of the Millstone Unit No. 3 Technical Specifications will 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.

DG Full-Load Rejection Test

NNECO is proposing to modify Surveillance Requirement 4.8.1.1.2.g.3 of the Millstone Unit No. 3 Technical Specifications by changing the acceptable transient voltage to 5000 volts from 4784 volts. The intent of the proposal is to permit the DG full load rejection tests to be conducted at conditions which simulate design basis conditions.

The proposed change does not alter the intent of the surveillance, does not involve any physical changes to the plant, does not alter the way any structure, system, or component functions, and does not modify the manner in which the plant is operated. As such, the proposed change to Surveillance Requirement 4.8.1.1.2.g.3 will not degrade the ability of the DGs to perform their intended safety function, and will not reduce the availability of the DGs.

The bases of Technical Specification 3/4.8, ''Electrical Power Systems,'' state that the operability of the AC and DC power systems and associated distribution systems ensure that sufficient power will be available to supply the safety related equipment required for safe shutdown and for the mitigation of transients. The proposed change to the surveillance requirement will increase the effectiveness of the full load rejection tests.

This will ensure the operability of the DGs. Operable DGs ensure that the assumptions for the bases of the Millstone Unit No. 3 Technical Specifications are not affected.

Pressure Test of the DG Fuel Oil System

NNECO is proposing to delete Surveillance Requirement 4.8.1.1.2.i.2 from the Millstone Unit No. 3 Technical Specifications. This surveillance requirement is redundant to the requirements of Surveillance Requirement 4.0.5 which invokes ASME Section XI. Additionally, the fuel oil system cannot be tested to the requirements of Surveillance Requirement 4.8.1.1.2.i.2 because the DG fuel oil tanks are vented to the atmosphere and this vent path cannot be isolated.

Millstone Unit No. 3 will include the DG fuel oil system pressure test as an augmented inspection within the Inservice Inspection program. Inspections will be performed in compliance with the requirement of the 1983 Edition of ASME Section XI, Table IWD–2500–1, ''Test and Examination Categories.'' Testing (i.e., a system hydrostatic test) in accordance with ASME Section XI will provide equivalent assurance of tank and piping integrity.

Conclusion

Based on the above, the proposed changes to Surveillance Requirements 4.8.1.1.2.g.3 and 4.8.1.1.2.i.2 of the Millstone Unit No. 3 Technical Specifications 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 amendment request involves no significant hazards consideration.

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

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: Phillip F. McKee.

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

Date of amendment request: January 18, 1995.

Description of amendment request: The proposed changes to the technical specifications will increase the minimum required boron concentration in the boric acid tank (BAT) from 6300 ppm to 6600 ppm.

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

* * * The proposed changes do not involve an SHC because the changes would not:

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

The change affects the minimum required boron concentration in the BAT. Changes in the tank's boron concentration will not affect the probability of any plant accident.

An increase in the minimum BAT concentration of 6600 ppm was recommended by Westinghouse based on their Cycle 6 BORDER evaluation. The BORDER evaluation conservatively determines the ability to maintain shutdown margin when the plant is taken from an initial operating condition of Mode 1 or 2 to a final condition of Mode 5 or 6 using an assumed minimum BAT concentration. Therefore, the ability to maintain shutdown margin is assured and the change will not adversely affect the consequences of any plant accident.

2. Create the possibility of a new or different kind of accident from any Previously Analyzed.

The change conservatively increases the minimum required boron concentration in the BAT from 6300 ppm to 6600 ppm. There is no impact on the operability of plant systems or equipment. Therefore, the change does not create a malfunction that is different from those previously evaluated.

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

The proposed increase in the minimum boron concentration in the BAT provides conservatism in the calculated shutdown margin for Millstone Unit No. 3. The change does not adversely affect any equipment credited in the safety analysis. Also, the change does not adversely affect the probability or consequences of any plant accident, including the calculated PCT [peak clad temperature] or offsite doses. Therefore, there is no impact on the margin of safety as specified in the Technical Specifications.

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

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

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: 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: January 10, 1995.

Description of amendment requests: The proposed amendments would revise the Prairie Island Event Monitoring Instrumentation Technical Specifications and associated Bases to conform to Standard Technical Specifications for post-accident monitoring.

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.

The primary purpose of post accident monitoring instrumentation is to display plant variables that provide information to the control room operators during accident situations. Plant instrumentation was evaluated for importance for this function when Regulatory Guide 1.97 [''Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident''] classifications were determined. The Prairie Island Regulatory Guide 1.97 classification of instruments was previously approved by the NRC on October 18, 1985. This amendment request proposes to base Prairie Island Technical Specifications on the results of the Regulatory Guide 1.97 evaluation in accordance with the guidance of the industry standard.

Revising the allowed outage time for these instruments will not significantly increase the probability or consequences of an accident since these instruments do not initiate automatic actions, there are available backup indications and the probability of an event requiring these instruments to be operable is very low.

Therefore, the probability or consequences of an accident previously evaluated are not affected by any of the proposed amendments.

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

The license amendment request proposes to add instruments to the Technical Specifications which have been previously determined to be important for post accident monitoring, and to remove instruments from Technical Specifications which have been previously determined to be less important for post accident monitoring. This amendment ensures the control room operators are provided with the instrumentation required to properly manage an accident situation.

Therefore, based on the above considerations, the possibility of a new or different kind of accident from any accident previously evaluated would not be created.

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

The post accident monitoring functions do not initiate any automatic actions. The instrumentation to be added to the Event Monitoring Instrumentation Table was previously recognized through the Regulatory Guide 1.97 evaluation process as important for post accident monitoring and would be relied upon if there were an event without this license amendment. Instrumentation to be removed from Technical Specifications was previously recognized to be less important and would not be relied upon very much in an event. Overall, with the trade-off of adding and deleting instrumentation, the margin of safety will not be significantly affected.

The proposed license amendment will increase the allowed outage time for most of the instruments. Again, these instruments do not provide automatic actions, they provide indications for monitoring post accident conditions. All of the instruments have backup or corroborating indications which could be relied upon if the Technical Specifications instruments were inoperable. Also, an event requiring use of these instruments has a very low probability. For these reasons the proposed changes in allowed outage time will not result in a significant reduction in the margin of safety.

For these same reasons, the proposed changes in radiation instrument surveillance requirements will not significantly reduce the margin of safety.

Overall, a significant reduction in the margin of safety would not result from this license amendment.

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.

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

Date of amendment request: January 25, 1995.

Description of amendment request: The proposed Technical Specification change would replace a specific requirement for the frequency of Type A tests with a general requirement to perform Type A tests. The proposed amendment would change Surveillance Requirement 4.6.1.2.a. Specifically, the change would require the performance of Type A tests (overall containment integrated leak rate tests (ILRTs)) at intervals as specified in 10 CFR 50, Appendix J, instead of on a specific schedule for performance of ILRTs of ''40 plus or minus 10 months.''

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

A. The change does not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)) because the proposed change merely replaces a prescriptive schedule for performing ILRTs with a requirement to conduct the ILRTs on a schedule consistent with the Commission's regulations. The change does not alter the methodology, frequency, or acceptance criteria for ILRTs, does not affect the design basis of the containment, and does not change the post-accident response of the containment.

B. The change does not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because the change does not affect the manner by which the facility is operated and does not make any changes to existing plant structures, systems, or components. The proposed change merely replaces a prescriptive schedule for performing ILRTs with a requirement to conduct the ILRTs on a schedule consistent with the Commission's regulations.

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

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

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

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

NRC Project Director: Phillip F. McKee.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendments request: December 7, 1994.

Description of amendments request: The amendments would provide a permanent voltage-based steam generator tube repair criteria for both units. This criteria is based on the guidance contained in the NRC Proposed Generic Communication (Generic Letter 94–XX), ''Voltage-Based Repair Criteria for the Repair of Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking,'' that was issued for public comment in the **Federal Register** (59 FR 41520) on August 12, 1994. The licensee's submittal also includes responses to and identifies exceptions taken to the draft Generic Letter. The significant exceptions are: (1) The requirement to reinspect all tubes if bobbin probe wear exceeds 15%; (2) the $1{\times}10^{-2}$ limit on the calculated conditional burst probability; and (3) the need to pull additional steam generator tubes to evaluate the current condition of the steam generator tubes. In addition, the operational leakage requirement for Unit 2 will be modified to reduce the total allowable primary-to-secondary leakage for any steam generator from 500 gallons per day (gpd) to 150 gallons per day.

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

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

Testing of model boiler specimens for free standing tubes at room temperature conditions shows burst pressures as high as approximately 5000 psi for indications of outer diameter stress corrosion cracking with voltage measurements as high as 26.5 volts. Burst testing performed on pulled tubes with up to 7.5 volt indications show burst pressures in excess of 5900 psi at room temperature. As stated earlier, tube burst criteria are inherently satisfied during normal operating conditions by the presence of the tube support plate. Furthermore, correcting for the effects of temperature on material properties and minimum strength levels (as the burst testing was done at room temperature), tube burst capability significantly exceeds the R.G. [Regulatory Guide] 1.121 criterion requiring the maintenance of a margin of 1.43 times the steam line break pressure differential on tube burst if through-wall cracks are present without regard to the presence of the tube support plate. Considering the existing data base, this criterion is satisfied with bobbin coil indications with signal amplitudes over twice the 2.0 volt voltage-based repair criteria, regardless of the indicated depth measurement. This structural limit is based on a lower 95% confidence level limit of the data. The 2.0 volt criterion provides an extremely conservative margin of safety to the structural limit considering expected growth rates of outside diameter stress corrosion cracking at Farley. Alternate crack morphologies can correspond to a voltage so that a unique crack length is not defined by a burst pressure to voltage correlation. However, relative to expected leakage during normal operating conditions, no field leakage has been reported from tubes with indications with a voltage level of under 7.7 volts for 3/4 inch tube which correlates to 10 volts for 7/8 inch tubing (as compared to the 2.0 volt proposed voltage-based tube repair limit). Thus, the proposed amendment does not involve a significant increase in the probability or consequences of an accident.

Relative to the expected leakage during accidents (sic) condition loadings, the accidents that are affected by primary-to-secondary leakage and steam release to the environment are Loss of External Electrical Load and/or Turbine Trip, Loss of All AC Power to Station Auxiliaries, Major Secondary System Pipe Failure, Steam Generator Tube Rupture, Reactor Coolant Pump Locked Rotor, and Rupture of a Control Rod Drive Mechanism Housing. Of these, the Major Secondary System Pipe Failure is the most limiting for Farley in considering the potential for off-site doses. The offsite doses analyses for the other events which model primary-to-secondary leakage and steam releases from the secondary side to the environment assume that the secondary side remains intact. The steam generator tubes are not subjected to a sustained increase in differential pressure, as is the case following a steam line break event. This increase in differential pressure is responsible for the postulated increase in leakage and associated offsite doses following a steam line break event. In addition, the steam line break event results in a bypass of containment for steam generator leakage. Upon implementation of the voltage-based repair criteria, it must be verified that the expected distributions of cracking indications at the tube support plate intersections are such that primary-to-secondary leakage would result in site boundary dose within the current licensing basis. Data indicate that a threshold voltage of 2.8 volts could result in through-wall cracks long enough to leak at steam line break conditions. Applications of the proposed repair criteria requires that the current distribution of a number of indications versus voltage be obtained during the refueling outages. The current voltage is then combined with the rate of change in voltage measurement and a voltage measurement uncertainty to establish an end of cycle voltage distribution and, thus, leak rate during steam line break pressure differential. The leak rate during a steam line break is further increased by a factor related to the probability of detection of the flaws. If it is found that the potential steam line break leakage for degraded intersections planned to be left in service coupled with the reduced specific activity levels allowed result in radiological consequences outside the current licensing basis, then additional tubes will be plugged or repaired to reduce steam line break leakage potential to within the acceptance limit. Thus, the consequences of the most limiting design basis accident are constrained to present licensing basis limits.

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

Implementation of the proposed voltage-based tube support plate elevation steam generator tube repair criteria does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism that could result in an accident outside of the region of the tube support plate elevations. Neither a single or multiple tube rupture event would be expected in steam generator in which the repair criteria have been applied during all plant conditions. The bobbin probe signal amplitude repair criteria are established such that operational leakage or excessive leakage during a postulate steam line break condition is not anticipated. Southern Nuclear has previously implemented a maximum leakage limit of 140/150 gpd (Unit 1/ Unit 2) per steam generator. The R.G. 1.121 criterion for establishing operational leakage limits that require plant shutdown are based upon leak-before-break considerations to detect a free span crack before potential tube rupture. The 140/150 gpd limit provides for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. R.G. 1.121 acceptance criteria for establishing operating leakage limits are based on leak-before-break considerations such that plant shutdown is initiated if the leakage associated with the longest permissible crack is exceeded. The longest permissible crack is the length that provides a factor safety of 1.43 against bursting at steam line break pressure differential. A voltage amplitude of approximately 9 volts for typical outside diameter stress corrosion cracking corresponds to meeting this tube burst requirement at the 95% prediction interval on the burst correlation. Alternate crack morphologies can correspond to a voltage so that a unique crack length is not defined by the burst pressure versus voltage correlation. Consequently, typical burst pressure versus throughwall crack length correlations is used below to define the ‘‘longest permissible crack’’ for evaluating operating leakage limits.

The single through-wall crack lengths that results in tube burst at 1.43 times steam line break pressure differential and steam line break conditions are about 0.53 inch and 0.84 inch, respectively. Normal leakage for these crack lengths would range from about 0.4 gallons per minute to 4.5 gallons per minute, respectively, while lower 95% confidence level leak rates would range from about 0.06 gallons per minute to 0.6 gallons per minute, respectively.

An operating leak rate of 140/150 gpd per steam generator has been implemented. This leakage limit provides for detection of 0.4 inch long cracks at nominal leak rates and 0.6 inch long cracks at the lower 95% confidence level leak rates. Thus, the 140/150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for steam line break conditions at leak rates less than 95% confidence level and for three times normal operating pressure differential at less than nominal leak rates.

Considering the above, the implementation of voltage-based plugging criteria will not create possibility of a new or different kind of accident from any previously evaluated.

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

The use of the voltage-based tube support plate elevation repair criteria is demonstrated to maintain steam generator tube integrity commensurate with the requirements of R.G. 1.121. R.G. 1.121 describes a method acceptable to the NRC staff for meeting GDCs [General Design Criteria] 2, 14, 15, 31, and 32 by reducing the probability of the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the criteria, even under the worst case conditions, the occurrence of outside diameter stress corrosion cracking at the tube support plant elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The most limiting effect would be a possible increase in leakage during a steam line break event. Excessive leakage during a steam line break event, however, is precluded by verifying that, once the criteria are applied, the expected end of cycle distribution of crack indications at the tube support plate elevations would result in minimal, and acceptable primary to secondary leakage during the event and, hence, help to demonstrate radiological conditions are less than an appropriate fraction of the 10 CFR 100 guideline.

The margin to burst for the tubes using the voltage-based repair criteria is comparable to that currently provided by existing technical specifications.

In addressing the combined effects of LOCA [loss-of-coolant accident] + SSE [safe shutdown earthquake] on the steam generator component (as required by GDC 2), it has been determined that tube collapse may occur in the steam generators at some plants. This is the case as the tube support plates may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to either the LOCA rarefaction wave and/or SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse.

There are two issues associated with steam generator tube collapse. First, the collapse of steam generator tubing reduces the RCS [reactor coolant system] flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase Peak Clad Temperature (PCT). Second, there is a potential the partial through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse or that short through-wall indications would leak at significantly higher leak rates than included in the leak rate assessments.

Consequently, a detailed leak-before-break analysis was performed and it was concluded that the leak-before-break methodology (as permitted by GDC 4) is applicable to the Farley reactor coolant system primary loops and, thus, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design basis of the plant. Excluding breaks in the RCS primary loops, the LOCA loads from the large branch line breaks were analyzed at Farley and were found to be of insufficient magnitude to result in steam generator tube collapse or significant deformation.

Regardless of whether or not leak-before-break is applied to the primary loop piping at Farley, any flow area reduction is expected to be minimal (much less than 1%) and PCT margin is available to account for this potential effect. Based on analyses' results, no tubes near wedge locations are expected to collapse or deform to the degree that secondary to primary in-leakage would be increased over current expected levels. For all other steam generator tubes, the possibility of secondary-to-primary leakage in the event of a LOCA + SSE event is not significant. In actuality, the amount of secondary-to-primary leakage in the event of a LOCA + SSE is expected to be less than that previously allowed, i.e., 500 gpd per steam generator. Furthermore, secondary-to-primary in-leakage would be less than primary-to-secondary leakage for the same pressure differential since the cracks would tend to tighten under a secondary-to-primary pressure differential. Also, the presence of the tube support plate is expected to reduce the amount of in-leakage.

Addressing the R.G. 1.83 considerations, implementation of the tube repair criteria is supplemented by 100% inspection requirements at the tube support plate elevations having outside diameter stress corrosion cracking indications, reduced operating leakage limits, eddy current inspection guidelines to provide consistency in voltage normalization, and rotating pancake coil inspection requirements for the larger indications left in service to characterize the principle degradation mechanism as outside diameter stress corrosion cracking.

As noted previously, implementation of the tube support plate elevation repair criteria will decrease the number of tubes that must be taken out of service with tube plugs or repaired. The installation of steam generator tube plugs or tube sleeves would reduce the RCS flow margin, thus implementation of the voltage-based repair criteria will maintain the margin of flow that would otherwise be reduced through increased tube plugging or sleeving.

Considering the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety as defined in the Final Safety Analysis Report or any bases of the plant Technical Specifications.

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: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201

NRC Project Director: William H. Bateman.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendments request: January 9, 1995.

Description of amendments request: The requested changes to the Technical Specifications (TS) would implement the recommended changes from Generic Letter 93–05, ''Line Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation.'' Specifically, the amendments would implement TS changes corresponding to the following GL 93–05 line-item improvement issues: Control Rod Movement Test for Pressurized Water Reactors, Radiation Monitors, Surveillance of Boron Concentration in the Accumulator/Safety Injection/Core Flood Tank, Containment Spray System, Hydrogen Recombiner, and Special Test Exemptions.

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 do not involve any change to the configuration or method

of operation of any plant equipment used to mitigate the consequences of an accident. The changes to the surveillance requirements will result in an overall improvement in plant safety by reducing the likelihood of plant trips and subsequent challenges to safety systems, decreasing equipment degradation due to excessive testing, reducing radiation exposure to plant personnel, increasing the availability of safety related equipment, and eliminating an unnecessary burden on plant personnel. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve any change to the configuration or method of operation of any plant equipment used to mitigate the consequences of an accident. The relaxation of surveillance tests curtails the excessive amount of testing that increases wear on the equipment and reduces the likelihood of plant trips and subsequent challenges to safety systems. The relaxation also increases the availability of safety related equipment. Accordingly, no new failure modes have been defined for any plant system or component important to safety nor has any new limiting failure been identified as a result of the proposed changes. 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 changes do not involve a significant reduction in a margin of safety. The proposed changes eliminate an unnecessary burden without compromising protection for public health and safety. The proposed changes were generically analyzed by the NRC as part of a comprehensive study and presented in NUREG–1366 ''Improvement to Technical specifications (sic) Surveillance Requirements.'' The NRC concluded that while some testing at power is essential to verify equipment and system operability, safety can be improved, equipment degradation decreased, and unnecessary personnel burden relaxed by reducing the amount of testing at power. SNC has analyzed plant operations and made a comparison with the criteria stated in NUREG–1366 for the line-item improvements contained in this request and has found the NUREG–1366 basis to be consistent with the Farley design and operation experience. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302.

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Project Director: William H. Bateman.

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: December 6, 1994.

Description of amendment request: The proposed change to Technical Specification 3/4.1.3.2 will delete Surveillance Requirement (SR) 4.1.3.2.2, that presently requires, every 31 days, the movement of at least 2% of its height for each Axial Power Shaping Rod not fully withdrawn. The proposed amendment would also change the surveillance intervals for the following Technical Specifications (TS) in accordance with the guidance of Generic Letter 93–05, ''Line Item Technical Specifications Improvements to Reduce Surveillance Requirements For Testing During Power Operation,'' and NUREG–1366, ''Improvements to Technical Specifications Surveillance Requirements:''

1. TS 4.1.3.2 for the Movable Control Assemblies ''Group Height—Safety and Regulating Rod Groups,'' will relax testing requirements from at least once every 31 days to every 92 days.

2. TS 4.4.6.2, for ''Operational Leakage,'' relaxes the requirement to leakage test RCS pressure isolation valves prior to MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours to whenever the plant has been in COLD SHUTDOWN for 7 days.

3. SR 4.5.2.c.2 for TS 4.5.2, ''ECCS Subsystems—Tavg equal to or greater than 280° F,'' relaxes the inspection requirements for ensuring no debris in containment from ''at the completion of each containment entry'' to ''at least once daily.''

4. TS 4.6.2.1.d, for the ''Containment Spray System,'' relaxes the SR to perform an air or smoke flow test through the spray header and nozzles from once per 5 years to once per 10 years.

5. TS 4.10.4.2 for ''Special Test Exceptions Shutdown Margin'' relaxes the SR interval for testing rod insertion capability prior to reducing shutdown margin from 24 hours to 7 days.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the NRC 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 change does not involve a significant increase in the probability of an accident previously evaluated nor does it involve a significant increase in the consequences of an accident previously evaluated because no change is being made to any accident initiator and no accident conditions or assumptions used in evaluating the radiological consequences of an accident are changed. Relaxation of surveillance requirements is in accordance with GL 93–05, NUREG–1366, and is compatible with plant operating experience. Deletion of SR 4.1.3.2 is consistent with NUREG–1430, ''Improved Standard Technical Specifications for B&W Plants.'' No credit is taken in any accident analysis or mitigation requirements for the Axial Power Shaping Rod Group.

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

The proposed changes do not create the possibility of any new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by these proposed changes. Relaxation of SRs as discussed in GL 93–05 was evaluated as reducing equipment degradation with no increase in safety consequences consistent with the maintenance of plant specific reliability of the equipment and systems affected. Deletion of the SR to move the Axial Power Shaping Rod Group does not affect the requirement to verify rod position, and there is no credit taken for movement of these rods to mitigate an accident.

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

The changes do not involve a significant reduction in the margin of safety, because the proposed changes affect only surveillance requirements, do not affect the function of the components and systems involved, and do not decrease the estimated equipment or system reliability.

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

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

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

NRC Project Director: Leif J. Norrholm.

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: December 6, 1994.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 4.0.5, ''Applicability'' and its associated Bases; TS 3/4.1.2.3, ''Reactivity Control Systems—Makeup Pump—Shutdown; TS 3/4.1.2.4, ''Reactivity Control Systems—Makeup Pump—Operating; TS 3/4.1.2.6, Reactivity Control Systems—Boric Acid Pump—Shutdown; and TS 3/4.1.2.7, ''Reactivity Control System—Boric Acid Pumps—Operating.'' The proposed change would replace the specific monthly surveillance requirements associated with the makeup pumps and boric acid pumps with a surveillance requirement referencing TS 4.0.5, which references Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code for quarterly pump testing requirements. The proposed change to TS 4.0.5 and its associated Bases would revise the requirement regarding the NRC's approval of relief requests to be in accordance with the NRC Staff's recommendation contained in NUREG–1482, ''Guidelines for Inservice Testing at Nuclear Power Plants.'' Additionally, TS 4.0.5.a.2 which describes historical requirements for inservice inspection and testing would be deleted and TS 4.0.5.a.1 would be renumbered as TS 4.0.5.a.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the NRC Staff has performed an 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.

Operation of the Davis-Besse Nuclear Power Station, Unit No. 1, in accordance with these changes, would not involve a significant increase in the probability of an accident previously evaluated because no accident initiators, conditions, or assumptions are affected by the proposed changes to replace the specific monthly surveillance requirements for the makeup and boric acid pumps with surveillance requirements referencing TS 4.0.5 (ASME Boiler and Pressure Vessel Code Section XI requirements) and to delete wording regarding NRC approval of relief requests. The changes do not involve a significant increase in the consequences of an accident previously evaluated, because no accident conditions or assumptions are affected that would increase the radiological consequences of a previously evaluated accident.

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

The proposed changes do not result in any new accident initiators nor do they alter any accident scenarios. The changes do not create the possibility of a different kind of accident from any accident previously evaluated, because the surveillance requirements for the makeup and boric acid pumps only affect the testing of existing components, systems, and functions, and do not introduce any new requirements.

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

The proposed changes do not reduce or adversely affect the capabilities or reliability of any plant structures, systems or components. Relaxation of the surveillance testing interval for the boric acid and makeup pumps and modifying the testing requirements is consistent with previous NRC guidance.

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

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

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

NRC Project Director: Leif J. Norrholm.

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

Date of amendment request: January 13, 1995.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) by relocating Tables 3.3–2, ''Reactor Trip System Instrumentation Response Times,'' and 3.3–5, ''Engineered Safety Features Response Times,'' to FSAR Chapter 16, Section 16.3. The Bases discussion specific to Table 3.3–5 would also be relocated to FSAR Section 16.3.

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

The proposed revision does not involve a significant hazards consideration because operation of Callaway Plant with this change would not:

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

Overall protection system performance will remain within the bounds of the accident analyses documented in FSAR Chapter 15, WCAP–10961–P, and WCAP–11883 since no changes to the response times or measurement interval are proposed.

The RTS and ESFAS will continue to function in a manner consistent with the above analysis assumptions and the plant design basis. As such, there will be no degradation in the performance of nor an increase in the number of challenges to equipment assumed to function during an accident situation.

These Technical Specification revisions do not involve any hardware changes nor do they affect the probability of any event initiators. There will be no change to normal plant operating parameters or accident mitigation capabilities. Therefore, there will be no increase in the probability or consequences of any accident occurring due to these changes.

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

As discussed above, there are no hardware changes associated with these Technical Specification revisions nor are there any changes in the method by which any safety-related plant system performs its safety function. The normal manner of plant operation is unaffected.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of these changes. Therefore, the possibility of a new or different type of accident is not created.

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

No response time changes are proposed in this amendment application; only the document where these limits are listed will be changed. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on DNBR limits, F_Q, F-delta-H, LOCA PCT, peak local power density, or any other margin of safety.

Based upon the preceding information, it has been determined that the proposed changes to the Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10CFR50.92(C) [sic] and do not involve a significant hazards consideration.

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

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

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

NRC Project Director: Leif J. Norrholm.

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

Date of amendment request: December 8, 1994.

Description of amendment request: The proposed amendment would change Standby Gas Treatment Power Supply Requirements during refueling operations.

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:

SGTS [Standby Gas Treatment System] DURING REFUELING OPERATIONS (Specification 3.7.B.1, 3.7.B.3)

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The Standby Gas Treatment System (SGTS) is not the initiator of any accident. SGTS may be required to operate for a design basis loss of coolant accident or for a refueling accident in order to mitigate the consequences of said accident by providing a filtered exhaust path to minimize the potential release of radioactive material to the environs. The proposed amendment does not reduce or change the operational requirements for the SGTS for an accident. The proposed amendment now clearly defines the operability requirements during refueling conditions. The proposed amendment further requires the availability of a second auxiliary power supply in the event that an Emergency Diesel Generator (EDG) is out of service during refueling operations, not currently required. We conclude, therefore, that the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The SGTS is not an accident initiator, therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in a margin of safety. The proposed amendment requires the availability of a second auxiliary power supply in the event that an EDG is out of service during refueling operations, not currently required. Maintaining availability of a specific reliable auxiliary electrical power source as an alternative to an EDG in this mode provides assurance that SGTS can, if required, be operated without placing undue constraints on EDG availability and represents an enhancement that increases a margin of safety. We conclude, therefore, that the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above discussion, we have determined that this change does not constitute a significant hazards consideration as defined in 10CFR50.92(c).

LABORATORY CARBON SAMPLE ANALYSIS (Specification 3.7.B.2.b)

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The Standby Gas Treatment System (SGTS) is not the initiator of any accident. SGTS may be required to operate for a design basis loss of coolant accident or for a refueling accident in order to mitigate the consequences of said accident by providing a filtered exhaust path to minimize the potential release of radioactive material to the environs. The proposed amendment does not reduce or change the operational requirements for the SGTS for an accident. The proposed amendment now clearly defines the operability requirements during the interval between sample removal and completion of laboratory analysis.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The SGTS is not an accident initiator, therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in a margin of safety. The proposed change does not reduce the requirements or acceptance criteria for sampling, testing or analysis. The proposed change only incorporates into the specification an existing clarification which addresses the determination of operability during the time between sample removal and completion of laboratory analysis. The change provides an explicit time limit consistent with current regulatory criteria for completion of analyses.

Based on the above discussion, we have determined that this change does not constitute a significant hazards consideration as defined in 10CFR50.92(c).

TORUS VENT MODE (Specification 4.7 B.2.c)

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The Standby Gas Treatment System (SGTS) is not the initiator of any accident. SGTS may be required to operate for a design basis loss of coolant accident or for a refueling accident in order to mitigate the consequences of said accident by providing a filtered exhaust path to minimize the potential release of radioactive material to the environs. The proposed amendment does not reduce or change the operational requirements for the SGTS for an accident.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The SGTS is not an accident initiator, therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in a margin of safety. The proposed change will incorporate into the specification an existing clarification. Use of the SGTS filters during Torus venting results in an insignificant flow through the filters. Further, maintaining humidity control prevents any adsorber degradation. Past sample testing on a six month calendar interval when 720 hours operating time has not accumulated has shown no detectable impact.

Based on the above discussion, we have determined that this change does not constitute a significant hazards consideration as defined in 10CFR50.92(c).

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

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, Vermont 05301.

Attorney for licensee: John A. Ritsher, Esquire, Ropes and Gray, One International Place, Boston, Massachusetts 02110–2624.

NRC Project Director: Walter R. Butler.

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.

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

Date of application for amendment: November 19, 1994.

Brief description of amendment: The proposed amendment would revise Section 3.10.8 and the associated Bases of the Indian Point Nuclear Generating Unit No. 3 Technical Specifications. Specifically, the proposed revision would reduce the maximum allowable control rod drop time from 2.4 to 1.8 seconds. The change would remove, for testing purposes, the allowance for a seismic event (0.6 seconds), which had been integral to the 2.4 second safety analysis basis. Since a seismic event cannot be simulated during the rod drop time test, the more conservative testing acceptance criteria value of 1.8 seconds is needed to ensure that the plant is within its design basis. This proposed revision will support control rod testing which is required during startup from the current outage.

Date of publication of individual notice in **Federal Register**: January 20, 1995 (60 FR 4203).

Expiration date of individual notice: February 21, 1995.

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

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.

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

Date of application for amendments: November 30, 1994.

Brief description of amendments: These amendments relocate Table 3.3–2, ''Reactor Protective Instrumentation Response Times,'' and Table 3.3–5, ''Engineered Safety Features Response Times,'' of TS 3/4.3.1 and 3/4.3.2, respectively, to the Palo Verde Updated Final Safety Analysis Report (UFSAR) in accordance with the guidance provided in Generic Letter 93–08. In addition, the amendments make administrative changes to two previous TS amendment requests to maintain consistency with the deletion of Tables 3.3–2 and 3.3–5. The amendments also delete an obsolete footnote on page 3/4 3–17 of the Palo Verde Unit 2's TS.

Date of issuance: February 3, 1995.

Effective date: February 3, 1995.

Amendment Nos.: 88, 75 and 59.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 4, 1995 (60 FR 496) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 3, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004.

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

Date of application for amendments: November 16, 1994.

Brief description of amendments: The proposed amendments change the Technical Specifications to revise the wording for the containment integrated leakage rate testing in Section 3/4.6.1.2 to make it consistent with the requirements of the BWR–4 Improved Standard Technical Specifications (NUREG–1433).

Date of issuance: January 26, 1995.

Effective date: January 26, 1995.

Amendment Nos.: 173 and 204.

Facility Operating License Nos. DPR–71 and DPR–62.

Date of initial notice in **Federal Register**: December 21, 1994 (59 FR 65810).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 26, 1995.

No significant hazards consideration comments received: No.

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

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: June 13, 1994, as supplemented on October 7, 1994.

Brief description of amendments: The amendments revise the administrative controls in Section 6 of the technical specifications (TS). The changes include: (1) a change to the submittal frequency of the Radiological Effluent Release Report from semiannually to annually; (2) changes to the Shift Technical Advisor (STA) description; (3) a clarification of the Shift Engineer responsibilities; and (4) several editorial changes.

Date of issuance: February 2, 1995.

Effective date: February 2, 1995.

Amendment Nos.: 69, 69, 59 and 59.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 26, 1994 (59 FR 53839).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 2, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: For Byron, the Byron Public Library, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Township Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

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

Date of application for amendment: May 17, 1993 as supplemented October 12, 1994.

Brief description of amendment: The amendment replaces License Condition 2.C.4, relating to the implementation and maintenance of the approved Fire Protection Program, in its entirety with a new License Condition. In conjunction, with this change, and in accordance with GL 86–10, Technical Specification provisions related to the Fire Protection Program are being deleted and placed in the Updated Final Safety Analysis Report.

Date of Issuance: February 1, 1995.

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

Amendment No.: 179.

Facility Operating License No. DPR–61. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 7, 1993 (58 FR 36432).

The October 12, 1994, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated February 1, 1995.

No significant hazards consideration comments received: No.

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

Duke Power Company, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina Date of application for amendments: August 25, 1994, as supplemented November 16, 1994.

Brief description of amendments: The amendments revise Technical Specification Table 3.3–4, by revising the ''Trip Setpoint'' and ''Allowable Value'' for the 4 kV bus undervoltage grid degraded voltage relays and the ''Allowable Value'' for the 4 kV undervoltage loss of voltage/loss of offsite power relays. This revision was submitted in response to a concern identified by the licensee in their Self-Initiated Technical Audit and during the electrical distribution system functional inspection team findings.

Date of issuance: January 20, 1995.

Effective date: To be implemented within 30 days from the date of issuance.

Amendment Nos.: 127 and 121.

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

Date of initial notice in **Federal Register**: October 12, 1994 (59 FR 51619).

The November 16, 1994, letter provided clarifying information that did not change the scope of the August 25, 1994, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 20, 1995.

No significant hazards consideration comments received: No.

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

Florida Power and Light Company, Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of application for amendment: July 28, 1994.

Brief description of amendment: This amendment revises Technical Specifications 3/4.4.13 to incorporate Low Temperature Overpressure Protection requirements similar to those recommended by the NRC staff via Generic Letter 90–06.

Date of Issuance: January 27, 1995.

Effective Date: January 27, 1995.

Amendment No.: 132.

Facility Operating License No. DPR–67: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 17, 1994 (59 FR 42341).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 27, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 3, 1994.

Brief description of amendments: The amendments relocate the requirements of Technical Specification 3/4.7.10, Area Temperature Monitoring, to section 16.3 of the VEGP Final Safety Analysis Report (FSAR). With this relocation to the FSAR, GPC plans to clarify the basis for areas to be monitored and modify these surveillance requirements. This change is in accordance with NUREG–1431, ''Standard Technical Specifications, Westinghouse Plants.''

Date of issuance: January 23, 1995.

Effective date: To be implemented within 30 days from the date of issuance.

Amendment Nos.: 83 and 61.

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

Date of initial notice in **Federal Register**: September 2, 1994 (59 FR 45735).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 23, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 13, 1992.

Brief description of amendment: The amendment changes the allowable primary-to-secondary leakage rate, as specified in License Condition 2.c.(8)2, from 0.1 gallons per minute (gpm) to 0.2 gpm.

Date of Issuance: January 31, 1995.

Effective date: January 31, 1995.

Amendment No.: 193.

Facility Operating License No. DPR–50. Amendment revises a License Condition.

Date of initial notice in **Federal Register**: October 14, 1992 (57 FR 47137).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated January 31, 1995.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: September 26, 1994.

Brief description of amendment: The amendment revised Technical Specification 3.5.C.1 and 3.5.C.4 to increase the minimum pressure at which the high pressure coolant injection system is required to be operable from 113 psig to 150 psig.

Date of issuance: January 25, 1995.

Effective date: January 25, 1995.

Amendment No.: 166.

Facility Operating License No. DPR–46. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 26, 1994 (59 FR 53841). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 25, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Auburn Public Library, 118 15th Street, Auburn, Nebraska 68305.

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

Date of amendment request: December 22, 1994.

Brief description of amendment: The amendment revised Technical Specification 1.0.J, definition of limiting conditions for operation, consistent with the guidance provided in NRC Generic Letter 87–09, ''Sections 3.0 and 4.0 of the Standard Technical Specifications on the Applicability of Limiting Conditions for Operation and Surveillance Requirements.''

Date of issuance: February 3, 1995.

Effective date: February 3, 1995.

Amendment No.: 168.

Facility Operating License No. DPR–46. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 3, 1995 (60 FR 153).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 3, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Auburn Public Library, 118 15th Street, Auburn, Nebraska 68305.

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

Date of application for amendment: July 21, 1994.

Brief description of amendment: The amendment revises Technical Specifications 2.2.2, 3.2.8, 4.2.8, and the associated Bases to reduce the number of reactor head safety valves required operable from 16 valves to 9 valves. The setpoints of the valve groups are unchanged by this amendment. The amendment requires testing of the safety valves in accordance with the approved NMP–1 Inservice Test Program.

Date of issuance: January 25, 1995.

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

Amendment No.: 152.

Facility Operating License No. DPR–63: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: August 31, 1994 (59 FR 45027).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 25, 1995.

No significant hazards consideration comments received: No.

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

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point

Nuclear Station, Unit 2, Oswego County, New York.

Date of application for amendment: October 28, 1994.

Brief description of amendment: The amendment revises Technical

Specification (TS) 1.7, ''CORE ALTERATION,'' to state that movement or replacement of incore instrumentation is not considered to be a CORE ALTERATION and that movement of control rods is not considered a CORE ALTERATION provided there are no fuel assemblies in the associated core cell. This amendment includes changes to TS 3/4.9.3, ''Control Rod Position,'' and associated Bases to be consistent with the revision to TS 1.7. TS 3/4.9.3 is being revised to require that all control rods be inserted only during loading of fuel assemblies into the core rather than during CORE ALTERATIONS. These changes are consistent with the NRC's, ''Improved Standard Technical Specifications,'' (NUREG–1434).

This amendment also revises Item 1.i.3) of TS Tables 3.3.2–1 and 4.3.2.1–1 to delete the requirement for Reactor Water Cleanup isolation due to actuation of the Standby Liquid Control System (SLCS) in OPERATIONAL CONDITION 5. License Amendment No. 48 issued on September 30, 1993, deleted the requirement for the SLCS to be OPERABLE in OPERATIONAL CONDITION 5; however, due to an oversight, Item 1.i.3) and associated notations were not deleted from TS Tables 3.3.2–1 and 4.3.2.1–1 as part of License Amendment No. 48. This amendment corrects that oversight.

Date of issuance: January 20, 1995.

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

Amendment No.: 61.

Facility Operating License No. NPF–69: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: November 23, 1994 (59 FR 60382).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 20, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: November 14, 1994.

Brief description of amendment: The amendment revises Technical Specification 4.5.1.e.2.e) to reduce the leak rate test pressure for the Automatic Depressurization System (ADS) nitrogen receiving tanks from 385 psig to 365 psig.

Date of issuance: January 31, 1995.

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

Amendment No.: 62.

Facility Operating License No. NPF–69: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: December 21, 1994 (59 FR 65817).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 31, 1995.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: January 14, 1994, as modified by letter dated October 17, 1994.

Description of amendment request: The amendment revises the Appendix A Technical Specifications (TS) to specify the composition of the Station Operation Review Committee (SORC) based on experience and expertise vice organizational position, to implement a Station Qualified Reviewer Program (SQRP), and to revise the time within which the Nuclear Safety Audit Review Committee (NSARC) must issue reports and minutes.

The amendment also incorporated a number of editorial changes to delete certain items that are no longer applicable; remove inconsistencies involving the names of systems, equipment and NSARC function, composition, and use of alternates; and correct the value for the reactor coolant system volume. Other editorial changes have been incorporated for document format consistency. The amendment affects the following: TS Sections 1.31, 3.3.3.6, 3.4.1.2, 4.6.3.2, 3.7.1.2, 3/4 10.6, 5.4.2, 6.3.1, 6.4, 6.7, and 6.8.1.4, and Table 4.3–1.

Date of issuance: January 26, 1995.

Effective date: January 26, 1995.

Amendment No.: 34.

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

Date of initial notice in **Federal Register**: May 25, 1994 (59 FR 27057) The licensee's letter dated October 17, 1994, provided clarification and minor revision to the application but does not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 26, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Exeter Public Library, 47 Front Street, 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: July 22, 1994.

Brief description of amendment: The amendment revises the Technical Specifications to incorporate a different setpoint and transient methodology for determining the maximum allowable power range neutron flux setpoint. These changes allow Millstone Unit 3 to operate with a reduced number of main steam-line safety valves at a reduced power level, as determined by the high flux setpoint.

Date of issuance: January 31, 1995.

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

Amendment No.: 102.

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

Date of initial notice in **Federal Register**: September 14, 1994 (59 FR 47171).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 31, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: October 25, 1994.

Brief description of amendments: These amendments add to the Susquehanna, Units 1 and 2, Technical Specifications, isolation signals to Table 3.6.3–1 for the containment isolation valves on the sample lines for the containment radiation monitoring and wetwell sample lines. This change is based on the licensee's design change for installation of a new CRM and wetwell sample system.

Date of issuance: January 31, 1995.

Effective date: January 31, 1995.

Amendment Nos.: 141 and 111.

Facility Operating License Nos. NPF–14 and NPF–22. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 7, 1994 (59 FR 63126). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 31, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 10, 1994, as supplemented by letter dated December 19, 1994.

Brief description of amendment: This amendment involves a one-time change affecting the Allowed Outage Time (AOT) for the Emergency Sevice Water (ESW) system, Residual Heat Removal Service Water (RHRSW) System, the Suppression Pool Cooling, the Suppression Pool Spray, and Low Pressure Coolant Injection modes of the Residual Heat Removal System, and Core Spray System to be extended from 3 and 7 days to 14 days during the Unit 2 refueling outage scheduled to begin in January 1995. This proposed extended AOT allows adequate time to install isolation valves and cross-ties on the ESW and RHRSW Systems to facilitate future inspections or maintenance.

Date of issuance: January 27, 1995.

Effective date: January 27, 1995.

Amendment No. 86.

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

Date of initial notice in **Federal Register**: July 20, 1994 (59 FR 37077). The December 19, 1994 letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 27, 1995.

No significant hazards consideration comments received: No.

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

Philadelphia Electric Company, Docket No. 50–353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Date of application for amendment: June 30, 1994.

Brief description of amendment: This amendment removes the controls for a remote shutdown system control valve and the primary containment isolation valves from TS Tables 3.3.7.4–1 and 3.6.3–1 respectively, as a result of eliminating the steam condensing mode of the Residual Heat Removal system.

Date of issuance: January 27, 1995.

Effective date: January 27, 1995.

Amendment No. 47.

Facility Operating License No. NPF–85. This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 17, 1994 (59 FR 42343).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 27, 1995.

No significant hazards consideration comments received: No.

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

Philadelphia Electric Company, Docket No. 50–353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania

Date of application for amendment: August 27, 1993, supplemented by letter dated November 17, 1993.

Brief description of amendment: The amendment allows an expanded operating domain for the Limerick Generating Station (LGS), Unit 2, resulting from the implementation of the Average Power Range Monitor—Rod Block Monitor Technical Specifications/Maximum Extended Load Line Limit Analysis. These improvements are a prerequisite for Power Rerate Program implementation at Limerick Generating Station, Unit 2.

Date of issuance: January 27, 1995.

Effective date: January 27, 1995.

Amendment No. 48.

Facility Operating License No. NPF–85. The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 13, 1993 (58 FR 52992). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 27, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: August 22, 1994.

Brief description of amendments: These amendments revise TS 3.1.5, ''Standby Liquid Control System,'' to remove the requirement for the standby liquid control system to be operable in OPERATIONAL CONDITION 5, Refueling, when any control rod is withdrawn and the TS definition of CORE ALTERATION to exclude control rod movement in a control cell that contains no fuel assemblies.

Date of issuance: January 27, 1995.

Effective date: January 27, 1995.

Amendment Nos. 87/49.

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

Date of initial notice in **Federal Register**: November 9, 1994 (59 FR 55881).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 27, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: September 28, 1993.

Brief description of amendment: The amendment revised Technical Specification (TS) Section 4.11.D to change the surveillance requirements for the Emergency Service Water System pumps. The change added pump flow rate requirements and tests the pumps in accordance with the licensee's Inservice Testing Program. The respective TS Bases were also revised.

Date of issuance: January 30, 1995.

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

Amendment No.: 223.

*Facility Operating License No. DPR–*59: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 24, 1993 (58 FR 62156).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 30, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: January 3, 1995 (TS 95–01).

Brief description of amendments: The amendments add a permissive statement to Surveillance Requirement 4.9.7.1 that will allow the auxiliary building bridge crane interlocks and physical stops to be defeated during implementation of the spent fuel pool storage capacity increase modification.

Date of issuance: January 24, 1995.

Effective date: January 24, 1995.

Amendment Nos.: 194 and 185.

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

Date of initial notice in **Federal Register**: January 9, 1995 (60 FR 2404) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 24, 1995.

No significant hazards consideration comments received: None.

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

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

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

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

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

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

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

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

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By March 17, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, 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, and to the attorney for the licensee.

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

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

Date of amendment request: July 26, 1994, as supplemented by letters dated December 27, 1994, and January 27, 1995.

Brief description of amendment: The amendment changed the Technical Specification Section 3/4.12.A to allow for increased flow capacity of the control room emergency filter system. By increasing the maximum allowed makeup capacity of this system, additional margin is provided for the positive pressurization of the control room envelope.

Date of issuance: January 27, 1995.

Effective date: January 27, 1995.

Amendment No.: 167.

Facility Operating License No. DPR–46. Amendment revised the Technical Specifications.

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

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

Local Public Document Room location: Auburn Public Library, 118 15th Street, Auburn, Nebraska 68305.

Attorney for licensee: Mr. G.D. Watson, Nebraska Public Power District, Post Office Box 499, Columbus, Nebraska 68602–0499.

NRC Project Director: William D. Beckner.

Dated at Rockville, Maryland, this 8th day of February 1995.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

Deputy Director, Division of Reactor Projects—III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 95–3629 Filed 2–14–95; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

[Rel. No. IC–20893; 811–3095]

Pacific American Fund; Notice of Application

February 9, 1995.

AGENCY: Securities and Exchange Commission (''SEC'').

ACTION: Notice of application for deregistration under the Investment Company Act of 1940 (the ''Act'').

APPLICANT: Pacific American Fund.

RELEVANT ACT SECTION: Section 8(f).

SUMMARY OF APPLICATION: Applicant seeks an order declaring that it has ceased to be an investment company.

FILING DATES: The application on Form N–8F was filed on January 11, 1995.

HEARING OR NOTIFICATION OF HEARING: An order granting the application will be issued unless the SEC orders a hearing. Interested persons may request a hearing by writing to the SEC's Secretary and serving applicant with a copy of the request, personally or by mail. Hearing requests should be received by the SEC by 5:30 p.m. on March 6, 1995, and should be accompanied by proof of service on the applicant, in the form of an affidavit or, for lawyers, a certificate of service. Hearing requests should state the nature of the writer's interest, the reason for the request, and the issues contested. Persons who wish to be notified of a