NUCLEAR REGULATORY COMMISSION

[Docket No. 50–244]

Rochester Gas and Electric Corporation; Ginna Nuclear Power Plant; Environmental Assessment and Finding of No Significant Impact

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an exemption from Facility Operating License No. DPR–18, issued to Rochester Gas and Electric Corporation (RG&E), (the licensee), for operation of the Ginna Nuclear Power Plant, located in Wayne County, New York.

Environmental Assessment

Identification of the Proposed Action

The proposed action would grant a one-time exemption from performing Type C tests for certain containment isolation valves (CIVs) during the 1995 refueling outage and extend the schedule required by 10 CFR Part 50, Appendix J, Section III.D.3, up to 1-month of the 2-year interval.

The proposed action is in accordance with the licensee's application for the exemption dated March 15, 1995.

The Need for the Proposed Action

The proposed action is requested on a one-time basis only to support the current refueling outage schedule. Requiring a plant shutdown before the next scheduled refueling outage in April 1996, soley to perform surveillance tests would cause an unnecessary thermal transient on the plant and could result in unnecessary exposure to personnel. The performance of the CIVs and the plant's overall containment integrity have been good. RG&E proposes to limit the exemption to exclude those valves: (1) On which maintenance has been performed; and (2) on those valves that have not demonstrated acceptable leakage rate testing during the past two leakage tests.

Environmental Impacts of the Proposed Action

The Commission has completed its evaluation of the proposed action and concludes that the proposed exemption would allow RG&E to conduct the local leak rate tests during the next refueling outage, an extension of up to 1 month. There will be no changes to the facility or the environment as a result of the exemption.

The change will not increase the probability or consequences of accidents, no changes are being made in the types of any effluents that may be released offsite, and there is no significant increase in the allowable individual or cumulative occupational radiation exposure. Accordingly, the Commission concludes that there are no significant radiological environmental impacts associated with the proposed action.

With regard to potential nonradiological impacts, the proposed action does involve features located entirely within the restricted area as defined in 10 CFR Part 20. It does not affect nonradiological plant effluents and has no other environmental impact. Accordingly, the Commission concludes that there are no significant nonradiological environmental impacts associated with the proposed action.

Alternatives to the Proposed Action

Since the Commission has concluded there is no measurable environmental impact associated with the proposed action, any alternatives with equal or greater environmental impact need not be evaluated. As an alternative to the proposed action, the NRC staff considered denial of the proposed action. Denial of the application would result in no change in current environmental impacts of the proposed action and the alternative action are similar.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the Final Environmental Statement for the Ginna Nuclear Power Plant.

Agencies and Persons Consulted

In accordance with its stated policy, on April 11, 1995, the staff consulted with the New York State official, Donna Ross, Acting State Liaison Officer of the New York Energy, Research, and Development Authority, regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

Based upon the environmental assessment, the Commission concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the Commission has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated March 15, 1995, which is available for public inspection at the Commission's Public Document Room, The Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Rochester Public Library, 115 South Avenue, Rochester, New York.

Dated at Rockville, Maryland, this 19th day of April 1995.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

Director, Project Directorate I–1, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 95–10207 Filed 4–25–95; 8:45 am]

BILLING CODE 7590–01–M

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from March 31, 1995, through April 14, 1995. The last biweekly notice was published on Wednesday, April 12, 1995 (60 FR 18621).

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 May 26, 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: March 20, 1995.

Description of amendment request: The licensee proposes a revision to Technical Specification (TS) 2.2.1, Reactor Trip System Instrumentation Setpoints, and to relocate cycle specific Overpower and Overtemperature Delta T trip setpoint parameters to the Core Operating Limits Report (COLR).

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

This change does not involve a significant hazards consideration for the following reasons:

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

The proposed change of relocating Overtemperature Delta T * * * and the Overpower Delta T * * * trip setpoint parameters to the COLR has no influence or impact to the probability or consequences of an accident. The revised TS will continue to implement the Reactor Trip System Instrumentation [Overtemperature Delta T] and [Overpower Delta T] setpoint limits through reference to the parameters in the COLR. In addition, the COLR is subject to the existing controls of TS 6.9.1.6, including the establishment of the parameter values using an NRC approved methodology. Given that this change administratively relocates the selected trip setpoint parameter values to another TS-controlled document, 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.

No safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. The limits are simply being relocated to another TS-controlled document. The TS will continue to require operation within the required limits as established per NRC approved methodologies. 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 amendment does not involve a significant reduction in the margin of safety.

Relocation of the Reactor Trip System Instrumentation [Overtemperature Delta T] and [Overpower Delta T] setpoint limits to the TS-controlled COLR has no effect on the trip system setpoints currently in force in TS 2.2.1. Future revisions to the trip setpoint parameters are governed by TS 6.9.1.6. TS 6.9.1.6 lists each TS that references values in the COLR and the NRC approved methodologies utilized in developing those values. Since this change is only an administrative relocation of the selected trip setpoint parameter values to another TS controlled document, 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: 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: David B. Matthews.

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

Date of amendment request: March 30, 1995.

Description of amendment request: The licensee proposes to revise the Emergency Diesel Generator (EDG) surveillance requirements contained in Technical Specification (TS) 4.8.1.1.2 to be consistent with NUREG–1431, Standard Technical Specifications for Westinghouse Plants, and to eliminate the need for duplicate EDG testing that has already been implemented to satisfy the requirements of the Station Blackout Rule and the Maintenance Rule.

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

This change does not involve a significant hazards consideration for the following reasons:

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

A failure of the Emergency Diesel Generator (EDG) is not an initiator for any previously evaluated FSAR Chapter 15 accident scenario. By committing to and implementing an EDG reliability program that satisfies the requirements of the Station Blackout Rule and the Maintenance Rule, the Shearon Harris Nuclear Power Plant (SHNPP) will continue to ensure that target EDG reliability and availability is being achieved by conducting appropriate monitoring, testing, and maintenance activities. This program will be developed and controlled as a Plant Operating Manual procedure and will incorporate industry, vendor, and TDI Owners Group recommendations. Therefore, with commensurate levels of testing and inspection in place to provide assurance that the EDGs will perform their intended safety function in the event of an accident, the proposed changes will have no effect on the probability or consequences of such an accident.

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

A failure of the EDG is not an initiator for any previously evaluated FSAR Chapter 15 accident scenario nor would the proposed changes to the EDG surveillance requirements result in the possibility of a new or different kind of accident from any accident previously evaluated. By committing to and implementing an EDG reliability program that satisfies the requirements of the Station Blackout Rule and the Maintenance Rule, SHNPP will continue to ensure that target EDG reliability and availability is being achieved by conducting appropriate monitoring, testing, and maintenance activities. This program will be developed and controlled as a Plant Operating Manual procedure and will incorporate industry, vendor, and TransAmerica Delaval Inc. Owners Group recommendations. Therefore, with commensurate levels of testing and inspection in place to provide assurance that the EDGs will perform their intended safety function in the event of an accident, the proposed changes would not increase the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes will not affect any parameters which relate to the margin of safety as defined in the Technical Specifications or the FSAR. Testing, inspection and maintenance necessary to verify the EDGs' ability to perform their intended safety function will continue to be performed. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: 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: David B. Matthews.

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: March 24, 1995.

Description of amendment request: The proposed amendments would acknowledge the acceptability of performing containment leakage rate testing in accordance with 10 CFR Part 50, Appendix J, and all approved 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 amendment does not involve a significant increase in the probability of occurrence or consequences of any accident previously evaluated.

The proposed changes to Technical Specifications to add an allowance to test in accordance with approved exemptions to 10 CFR 50 Appendix J are administrative in nature and will not affect any accident initiators or precursors. 10 CFR 50 Appendix J provides the requirements to periodically test the primary reactor containment. The objective of these requirements is to ensure that leakage from the primary reactor containment structure and systems and components that penetrate the containment is maintained below the limits established for containment leakage. The performance of periodic integrated leakage rate testing (Type A) and local penetration testing (Type B and C) during containment life provides a current assessment of potential leakage from containment during accident conditions.

10 CFR 50.12 allows the Commission to grant specific exemptions to the requirements of 10 CFR 50 Appendix J when those exemptions are authorized by law, will not present undue risk to the public, and are consistent with the common defense and security. In addition, special circumstances must exist as described in Section 50.12. Since all exemptions to 10 CFR 50 Appendix J receive NRC review and approval prior to being implemented, all containment leakage rate testing will continue to be performed in accordance with NRC approved methodologies when relying upon the allowance that is added to the Technical Specifications by the proposed amendment. The proposed changes are consistent with the requirements provided in NUREG–1431, ''Standardized Technical Specifications, Westinghouse Plants'' which has been approved by the NRC.

The proposed changes will not affect any accident initiators or precursors and will not change or alter the design assumptions for the systems used to mitigate the consequences of an accident. The proposed changes do not involve the addition of any new or different type of equipment, nor do they involve the operation of equipment required for safe operation of the facility in a manner different from those addressed in the UFSAR. There are no changes to parameters governing plant operation as a result of the proposed changes. The results and conclusions in the Zion Updated Final Safety Analysis Report (UFSAR) are unaffected by this proposed License Amendment.

Based on the previous discussion, the proposed changes do not involve a significant increase in the probability of occurrence or consequences of any accident previously evaluated.

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

The proposed changes to Technical Specifications to add an allowance to perform containment leakage rate testing in accordance with approved exemptions to 10 CFR 50 Appendix J are administrative in nature and do 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 analyses. The proposed changes may only affect the methods used to perform containment leakage rate testing while in a shutdown condition. 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.

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

Given the proposed changes to Technical Specifications, testing would be allowed in accordance with approved exemptions to Appendix J. Exemptions are allowed by the Commission in accordance with 10 CFR 50.12 when it is shown that the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. In addition, special circumstances must exist.

The proposed changes will not impact any margin of safety and testing in accordance with approved exemptions will not involve a significant reduction in a level of safety since containment leakage testing is performed while in a shutdown condition. In addition, it is likely that any test methodology that significantly reduces a margin of safety would not be approved by the NRC.

The ability to safely shut down the operating unit and 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.

Consumers Power Company, Docket No. 50–155, Big Rock Point Plant, Charlevoix County, Michigan

Date of amendment request: December 15, 1994.

Description of amendment request: The proposed amendment would revise Technical Specification 11.3.1.5 *ACTION* a. to eliminate the need to demonstrate that the actuation circuitry of the unaffected reactor depressurization system channels is operable. In addition, an editorial change correcting a typographical error is also proposed.

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

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

The proposed change will eliminate the probability of a subsystem failure caused by additional testing (which unnecessarily introduces the potential for human and equipment problems), therefore eliminating the probability that the facility would have to be challenged and brought to the SHUTDOWN condition within 12 hours and to the COLD SHUTDOWN condition within the following 24 hours.

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

The proposed change does not alter the plant configuration, systems, components, or operation; and does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change is expected to eliminate unnecessary challenges to a safety system that has already been determined to be operable by routine surveillance testing; therefore contributing to the overall safe operation of the facility.

3. Will the proposed change involve a significant reduction in the margin of safety?

The RDS [Reactor Depressurization System] provides for both manual and automatic depressurization of the primary system to allow injection of the core spray following a small-to-intermediate size break in the primary system. This will allow core cooling with the objective of preventing excessive fuel clad temperatures. The design of the system is based on the specified initiation set points described in the Technical Specifications. Transient analysis demonstrated that these conditions result in adequate safety margins for both the fuel and the system pressure. The proposed change does not affect these setpoints, therefore the margin of safety is not changed.

In addition, the proposed editorial change to correct a typographical error is administrative in nature and, therefore, would have no effect on the three standards of 10 CFR 50.92 discussed above.

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: North Central Michigan College, 1515 Howard Street, Petoskey, Michigan 49770.

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

NRC Project Director: Cynthia A. Carpenter, Acting.

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

Date of amendment request: January 26, 1995, as supplemented March 9, 1995.

Description of amendment request: The proposed amendment would revise the technical specifications (TS) to increase the allowable nominal fuel enrichment from 4.2 to 5.0 weight percent for reload fuel assemblies. TS impose a limit on fuel enrichment of stored fuel assemblies to prevent inadvertent criticality. Presently, the Crystal River Unit 3 (CR3) TS specify a maximum enrichment of 4.5 weight percent for storage pool A and dry fuel (new fuel) storage racks, and 4.2 weight percent for fuel pool B. The licensee proposed to revise TS 3.7.15, 4.2, and 4.3, and associated TS bases to allow increasing the enrichment limits from 4.2 to 5.0 weight percent for the dry fuel storage racks and for A and B fuel pools. Additionally, a typographical error in TS 4.3.1.2.b will also be corrected.

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

An increase in fuel enrichment will not by itself affect the mixture of fission product nuclides. A change in fuel cycle design which makes use of an increased enrichment may result in fuel burnup consisting of a somewhat different mixture of nuclides. The effect in this instance is insignificant because:

a. The isotopic mixture of the irradiated assembly is relatively insensitive to the assembly's initial enrichment.

b. Most accident doses are such a small fraction of 10 CFR 100 limits, a large margin exists before any change becomes significant.

c. The change in Pu content which would result from an increase in burnup would produce more of some fission product nuclides and less of other nuclides. Small increases in some doses are offset by reductions in other doses. The radiological consequences of accidents are not significantly changed.

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

An unplanned criticality event will not occur as K_{eff} [effective neutron multiplication factor] will not exceed 0.95 with the maximum allowable enriched fuel in Pool A and Pool B, when flooded with unborated water, and K_{eff} will not exceed 0.98 in the new fuel storage racks assuming conditions of optimum hypothetical low density moderation. The new fuel storage racks have two rows of storage cells physically blocked to ensure reactivity limits are not exceeded. Administrative controls assure fuel is stored in configurations which meet the requirements of the safety analysis.

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

While the increased enrichment in Pool A, Pool B, and the dry storage racks may lessen the margin to criticality, this reduction is not significant because the overall safety margin is within NRC criteria of K_{eff} [less than or equal to] 0.95 (NRC Standard Review Plan, Section 9.1.2.)

Therefore, this amendment request satisfies the criteria specified in 10 CFR 50.92 for amendments which 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 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: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 32629.

Attorney for licensee: A. H. Stephens, General Counsel, Florida Power Corporation, MAC–A5D, P. O. Box 14042, St. Petersburg, Florida 33733.

NRC Project Director: David B. Matthews.

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

Date of amendment request: March 16, 1995.

Description of amendment request: The proposed amendment would revise Technical Specification 4.6.1.2, regarding the overall integrated containment leakage rate tests, so that it would reference 10 CFR Part 50, Appendix J directly, rather than paraphrase the regulation, and allow approved exemptions to the test frequency requirements. In addition, there is an associated proposed exemption, from the requirements of 10 CFR Part 50, Appendix J, to provide a one-time interval extension for the Unit 2 Type A test (containment integrated leak rate test) from the current scheduled 48 months to approximately 66 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, 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 specific to Unit 2 will provide a onetime exemption from the 10 CFR 50, Appendix J Section III.D.I.(a) leak rate test schedule requirement. This change will allow for a one-time test interval for Type A Integrated Leak Rate Tests of approximately 66 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. Experience at South Texas Project Unit 2 demonstrates that excessive containment leakage paths are local leakage detected by Type B and C Local Leak Rate Tests.

Administrative controls govern 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, a Local Leak Rate Test is performed to ensure acceptable leakage levels. Following a Local Leak Rate Test 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 South Texas Project Unit 2 has been in operation, it has successfully passed each Type A Integrated Leak Rate Test. Therefore, a one-time exemption increasing the interval for performing an Integrated Leak Rate Test results [sic] in a significant decrease in the confidence in the leak tightness of the containment structure. Therefore, this change does not involve a significant increase in the consequences of an accident previously evaluated.

The proposed amendment revised Technical Specification 4.6.1.2 to reference the testing frequency requirements of 10 CFR 50, Appendix J, and to state that Nuclear Regulatory Commission approved exemptions to the applicable regulatory requirements are permitted. This portion of the proposed change is applicable to Unit 1 and Unit 2. The current language of Technical Specification 4.6.1.2 paraphrases the requirements of Section III,D.I.(a) [sic] 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.I.(a) will require a submittal from Houston Lighting & Power under 10 CFR 50.12 and subsequent review and approval by the Nuclear Regulatory Commission prior to implementation.

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

Therefore, this proposed amendment 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 Unit 2 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 66 months for the Integrated Leak Rate Test. Because the test history of South Texas Project Unit 2 demonstrates 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 for Units 1 and 2 provides clarification to a specification that paraphrases a codified requirement.

Since the proposed change and amendment would not change the design, configuration or method of operation of the plant, they 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 Integrated Leak Rate Tests is to ensure that release of radioactive materials will be restricted to those leak paths and leak rates assumed in accident analyses. The relaxed schedule for Integrated Leak Rate Tests does not allow for relaxation of Type B and C Local Leak Rate Tests. Therefore, methods for detecting local containment leak paths and leak rates are unaffected by this proposed change. A one-time increase of the South Texas Project Unit 2 test interval does not leak to a significant probability of creating a new leakage path or increased leakage rates because the test history for Integrated Leak Rate Tests shows no failure during plant life. The margin of safety inherent in existing accident analyses is maintained.

The proposed Technical Specification amendment for Units 1 and 2 is administrative and clarifies the relationship between the requirements of Technical Specification 4.6.1.2, Appendix J, and any approved exemptions to Appendix J. It does not, in itself, change a safety limit, a Limiting Condition of Operation, or a surveillance requirement on equipment required to operate the plant. Nuclear Regulatory Commission approval of any proposed change or exemption to III.D.1.(a) of Appendix J will be required prior to implementation.

Therefore, this change and amendment 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 standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

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

Attorney for licensee: Jack R. Newman, Esq., Newman & Holtzinger, P.C., 1615 L Street, NW., Washington, DC 20036.

NRC Project Director: William D. Beckner.

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: March 10, 1995.

Description of amendment request: The proposed amendment would remove redundant Limiting Conditions of Operation and Surveillance Requirements for the containment hydrogen and oxygen monitors in the Technical Specifications (TS).

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. No physical changes will result from this amendment. This change deletes requirements that are redundant and unduly restrictive. The annual surveillance deleted by this amendment is redundant to the semi-annual surveillance required in Table 4.2–H. The Limiting Conditions for Operation are not changed by the proposed amendment.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. No physical changes will result from this amendment. Functional tests are performed on the hydrogen and oxygen analyzers semiannually as required in TS Table 4.2–H. Deleting the annual requirement for a functional test of the same equipment will not reduce the amount of testing performed or increase the possibility of degraded equipment being undetected.

3. The proposed amendment does not involve a significant reduction in a margin of safety. No physical changes will result from this amendment. The existing requirement for a semi-annual test of the hydrogen and oxygen analyzer in Table 4.2–H exceeds the requirements to be deleted in Section 3.7/4.7–H. The frequency of testing of the hydrogen and oxygen analyzers will not be reduced as a result of this 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 request involves no significant hazards consideration.

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

Attorney for licensee: Jack Newman, Kathleen H. Shea, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Project Director: Gail H. Marcus.

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: March 28, 1995.

Description of amendment request: The proposed amendment would revise and clarify Technical Specification Table 3.2–A that lists allowable out-of-service times and surveillance test intervals for instrumentation.

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 to TS Table 3.2–A will not significantly increase the probability or consequences of an accident previously evaluated. The changes do not alter the physical design or operation of the plant and serve to describe more accurately and clearly the actual logic configurations. The existing logic designs are in conformance with the Architect/Engineer's design documentation since plant startup. These changes will assure that the information in the tables is clearer and more consistent with the column headings of the table. The proposed changes do not affect assumptions contained in the plant safety analysis.

The Bases changes provide additional information about the logic arrangements as appropriate to identify unique or different logic configurations. Changes to the Allowed Outage Time (AOT) descriptions for the MSL Flow—High and MSL Tunnel Temperature—High provide clarification regarding application of the AOT to these logic arrangements, since multiple instrument channels provide input into multiple logic channels. This application conforms to the single failure criterion of the design basis (NEDO–10139, Compliance of Protection Systems to Industry Criteria: General Electric BWR Nuclear Steam Supply System, dated June 1970) and to the analytical basis for the TS (NEDC–31677P–A, Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation, dated July 1990).

2. The proposed changes to Table 3.2–A will not introduce a new or different kind of accident from any accident previously evaluated. The changes do not alter the physical design of the plant or affect any modes of operation. The proposed changes serve to clarify the existing information to better assure that the trip instrumentation will be maintained as assumed in the accident analyses contained in the Updated Final Safety Analysis Report.

3. The proposed changes do not involve a significant reduction in a margin of safety. Clarification of the logic arrangements in both Table 3.2–A and the TS Bases and how the AOT is applied does not affect the ability of the isolation logic to perform its intended function. No physical changes to the plant are being made as part of this 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 request involves no significant hazards consideration.

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

Attorney for licensee: Jack Newman, Kathleen H. Shea, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Project Director: Gail H. Marcus.

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

Date of amendment request: March 17, 1995.

Description of amendment request: The proposed amendment would defer performance of the Type A containment integrated leakage rate test until the next refueling outage.

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:

Per 10 CFR 50.92, a proposed change does not involve a significant hazards consideration if the change does not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated,
2. Create the possibility of a new or different kind of accident from any accident previously evaluated, or
3. Involve a significant reduction in a margin of safety.

Criterion 1

The Cook Nuclear Plant Type A test history provides substantial justification for the proposed test schedule. Three Type A tests were performed over a seven year period with successful results. The tests indicate that the Cook Nuclear Plant has a low leakage containment. In addition, there are no adverse trends in the results from the previous Types A, B, and C tests or visual inspections that indicate a gradual degradation of the containment boundary. Further, there are no structural modifications planned which would adversely affect the structural capability of the containment and that would be a factor in deferring the Type A test one refueling outage. Containment leak rate testing is not an initiator of any accident. The proposed interval extension does not affect reactor operations or the accident analysis and has no radiological consequences, except for the dose savings associated with not performing the test. There will be no changes to 10 CFR 100 dose limits or the control room dose limits. Extending the test interval will not increase the probability of a malfunction of equipment important to safety. Based on these considerations, it is concluded that the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2

The proposed change does not involve physical changes to the plant or changes in plant operating configuration. The proposed change only relaxes the schedular requirements for conducting one Type A test from the T/Ss and defers performance of the test one cycle. The purpose of the test is to provide periodic verification of the leak-tight integrity of the primary reactor containment, and systems and components which penetrate containment. The tests assure that leakage through containment and systems and components penetrating containment will not exceed the allowable leak rate values established in 10 CFR 50, Appendix J. Thus, it is concluded that the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3

The proposed change to the schedule for performing the Type A test does not reduce the margin of safety assumed in the accident analysis for any release of radioactive materials or reduce any margin of safety preserved by the technical specifications. The methodology, acceptance criteria, and the technical specification leak rate limits for the performance of the Type A test will not change. Type A tests will continue to be performed in accordance with 10 CFR 50, Appendix J and the applicable Cook Nuclear Plant Technical Specifications beginning in 1997. In addition, there are no adverse trends in the results from the previous Type A, B, and C tests or visual inspections that indicate a gradual degradation of the containment boundary. Based on these considerations, it is concluded that the change does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts and

Trowbridge, 2300 N Street, NW, Washington, DC 20037.

NRC Project Director: Cynthia A. Carpenter, Acting.

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

Date of amendment requests: March 31, 1995.

Description of amendment requests: The proposed amendments would modify the Containment Ventilation System Technical Specifications (and associated Bases) to allow limited containment purge operation in Modes 1, 2, 3, and 4 for pressure control, ALARA [as low as is reasonably achievable], and respirable air quality considerations.

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

Criterion 1

The purpose of this amendment is to allow flexibility in the use of the containment purge system during MODES 1, 2, 3, and 4. The use of this system during these modes of operation has previously been approved (Amendment No. 66). Therefore, this amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed change to the T/Ss does not affect the assumptions, parameters, or results of any UFSAR [Updated Final Safety Analysis Report] accident analysis. Based on the existing system design and demonstrated closure capability it is concluded that the proposed changes do not modify the response of the containment during a design basis accident. The proposed amendment does not add or modify any existing equipment. Based on these considerations, it is concluded that the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2

The proposed change does not involve physical changes to the plant or changes in the plant operating configuration. Thus, it is concluded that the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3

The margin for safety presently provided is not reduced by the proposed change. As discussed previously, the containment purge valves have been designed and demonstrated capable of closure against the dynamic forces resulting from a loss of coolant accident. The proposed amendment does not impact the ability of the purge valves to perform their intended function (i.e. achieve closure) in the event of an accident. Based on these considerations, it is concluded that the 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 requests involve no significant hazards consideration.

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

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

NRC Project Director: Cynthia A. Carpenter, Acting.

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

Date of amendment request: March 31, 1995.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to increase the as-found setpoint tolerance of the safety/relief valves (SRVs) from plus or minus 1% to plus or minus 3%. In addition, the proposed amendment (1) would allow the as-found condition of one SRV to be inoperable, (2) clarifies the 1325 psig safety limit wording, (3) increases the number of SRVs to be tested during each refueling outage, (4) makes editorial changes to reflect the TS changes, and (5) revises the bases for the applicable sections.

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

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

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

The safety function of the SRVs is to mitigate the effects of a RPV [reactor pressure vessel] overpressurization, therefore a failure to open until the upper setpoint limit (+3%) is reached cannot affect the probability of an accident. The lowest allowable limit (−3%) is still above normal operating pressure and therefore does not significantly increase the probability of an inadvertent opening.

Should the SRVs open in response to an RCS [reactor coolant system] overpressure event, opening of the SRVs below the nominal setpoints does not adversely affect the consequences of an accident. The fuel reload analysis demonstrates that actuation of five valves at or below 103% of nominal provides sufficient pressure reduction to maintain peak RCS pressure below the safety limit of 1375 psig and to maintain vessel steam space pressure below 1325 psig. The hydrodynamic loads on the SRV discharge pipe (i.e., tail pipe) and the torus remain within the design limits.

The performance of the high pressure systems; FWCI [feedwater coolant injection], SLC [standby liquid control] and IC [isolation condenser] remain acceptable. There is also no adverse impact on the operability of the APR [automatic pressure relief] system.

The SRV setpoints will continue to be required to be within [plus or minus] 1% prior to plant startup from a refueling outage. This ensures that the SRVs are restored to the optimal conditions at the start of each fuel cycle.

Therefore, increasing the ''as-found'' tolerance from [plus or minus] 1% to [plus or minus] 3% does not result in a significant increase in the probability or consequences of a previously analyzed accident.

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

Revising the acceptable as-found setpoint tolerance from [plus or minus] 1% to [plus or minus] 3% does not change the type of action that these valves are expected to perform, nor does it change the initial ''as-left'' requirements for the valves. Plant operating parameters have not changed. Therefore, this change cannot create the possibility of a new or different kind of accident.

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

The margin of safety established and stated in the Millstone Unit No. 1 Technical Specifications, is a peak RCS pressure of 1375 psig and a peak vessel steam space pressure of 1325 psig. While allowing the SRV setpoint tolerance to increase to [plus or minus] 3% would allow peak pressures from an MSIV [main steam isolation valve] closure event to approach that safety limit, the safety limit will not be exceeded. Therefore, this change does not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: 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.

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

Date of application for amendments: February 10, 1995.

Description of amendment request: The proposed changes provide for the correction of administrative errors made in the past during the processing of technical specification changes related to control room ventilation filter surveillance testing.

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

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because the changes are purely administrative and do not involve any physical changes to plant SSC [systems, structures, or components]. Therefore, these changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because the changes will not alter the plant or the manner in which the plant is operated. The changes do not allow plant operation in any mode that is not already evaluated in the safety analysis. The changes will not alter assumptions made in the safety analysis and licensing bases. Therefore, these changes will 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 because they are purely administrative and have no impact on any safety analysis assumptions.

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

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

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: March 15, 1994.

Description of amendment request: This amendment would reflect an exemption from 10 CFR Part 50, Appendix J, Section II.H.4, concerning the scope of Type 'C' testing on specified emergency core cooling system and reactor core isolation cooling containment isolation valves by revising Technical Specification Table 3.6.3–1, Primary Containment Isolation Valves. The subject valves on systems which terminate below the minimum water level of the suppression pool and are associated with closed systems would be tested using requirements of the American Society of Mechanical Engineers' Section XI Code.

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

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

The proposed change to the scope of Type 'C' testing for the subject valves does not affect the probability of the design basis accidents. The valves will continue to be maintained in an operable state, and in their current design configuration. There is no correlation between the scope of the Type 'C' testing and accident probability.

PP&L reviewed the postulated consequences of design basis events on primary containment isolation under the proposed change. GDC 50 design conformance states that the primary containment structure, including access openings, penetrations and the containment heat removal system, is designed so that the containment structure and its internal compartments can withstand, without exceeding the design leakage rate (1.0% per day), the peak accident pressure and temperature that could occur during any postulated LOCA.

For the purposes of considering the consequences of LOCAs under the proposed change, a single active failure of a CIV or a passive failure of the closed system were reviewed, within the limits of the existing licensing basis. Under the existing licensing basis, a pipe rupture of seismically qualified ECCS piping does not have to be assumed concurrent with the LOCA, except if it is a consequence of the LOCA. Consequential failures can be eliminated, since a LOCA inside containment is separated from the ECCS piping by the containment structure. Consequential failures of the ECCS piping from LOCA's outside containment are outside the Appendix J design considerations, although they are adequately addressed through the redundancy and separation of the ECCS design. A single active failure of the CIV, under the LOCA condition, can be accommodated since the closed and filled system piping remains as the leakage barrier. The ECCS passive failure criterion does require consideration of system leaks, but not pipe breaks, beyond the initiating LOCA. Pipe leakage, equivalent to the leakage from a valve or pump seal failure, should be considered at 24 hours or greater post-LOCA. The capability to make-up inventory to the suppression pool is adequate to ensure that postulated seat leakage and pipe leakage does not result in a condition that jeopardizes pool level. Make-up capability exists to the suppression pool via the Condensate Storage Tank and Spray Pond. Actions to make-up to the suppression pool are delineated in Emergency Operating Procedures.

Therefore, the proposal to eliminate the subject Type 'C' tests does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The acceptability of the proposed change to the scope of Type 'C' testing for the subject valves is based on maintaining the existing barriers to primary containment leakage, and ensuring that the suppression pool level is assured for 30 days during all design basis, post-accident modes of operation. By meeting these dual objectives, the plant response to the design basis events will be unchanged, and no new accident scenarios will be encountered. These two objectives are related, in that, the suppression pool inventory creates a passive barrier to primary containment atmospheric leakage for penetrations which are located below the minimum water level of the pool. The subject valve lines terminate below the minimum suppression pool water level.

The subject valves are all single isolation valves associated with lines that penetrate the primary containment, but are not connected directly to the primary containment atmosphere or the reactor coolant pressure boundary. The redundant isolation boundary for each of the affected valves is the closed system associated with the valve. This configuration is described in General Design Criteria (GDC) 57. The proposed exemption, and Technical Specification change, does not alter the configuration of these systems. The valves will continue to be tested and maintained to ensure their operability. The closed system piping meets PP&L's design conformance to GDC 56 and is verified via a 10CFR50 Appendix J Type 'A' test. The integrity of the closed systems is also monitored and controlled via Technical Specification 6.8.4.a.

The subject valves may be open, or change state, postaccident to support the design function of their associated ECCS systems (HPCI, Core Spray, RHR) or RCIC. The subject valves function as system valves during the periods when they are open or in an intermediate state, not as containment isolation valves. Reliance is placed on the suppression pool seal and the closed system piping to maintain the barrier between primary and secondary containment atmospheres.

Therefore, with the valve and closed system configuration unaffected by the proposed change, the existing barriers to primary containment atmospheric leakage are maintained, so long as the suppression pool level is ensured.

The suppression pool is designed and operated so that it is filled with water in accordance with Technical Specifications 3/4.5.3, ''Suppression Chamber,'' 3/4.6.2, ''Depressurization Systems—Suppression Chamber,'' and the associated Bases. The supply of water in the suppression pool is assured for 30 days during all design basis, post-accident modes of operation. Type 'C' leak rate testing has historically been performed on valves associated with lines that connect to the suppression pool. The acceptance criteria for combined leakage from these penetrations is 3.3 gpm. This leakage rate is at a level which ensures the 30 day post-accident suppression pool level. However, for the valves discussed in this change, seat leakage past the CIV is into a closed and filled system. Thus ''leakage'' from the suppression pool, past the CIV, is a function of closed system leakage.

As mentioned above, the integrity of the closed system piping is verified via a 10CFR50 Appendix J Type 'A' test and is monitored and controlled via Technical Specification 6.8.4.a. TS 6.8.4.a establishes a program to monitor and control leakage from systems located outside containment that could contain highly radioactive fluids during a serious transient or accident. This program applies to the ECCS systems and RCIC affected by the proposed change and ensures that leakage into secondary containment via packing, flanges, seals, etc., is controlled. Leakage from these systems, plus the Scram Discharge Volume, Reactor Water Clean-up, and PASS, has been found to be very low, and well below the 5 gpm limit established for these systems. Current leakage for Unit 1 is 0.14 gpm and for Unit 2, 0.043 gpm. The proposed change is not expected to contribute to higher levels of system leakage. Any leakage from these systems is processed via Standby Gas Treatment and the radwaste system to maintain ALARA and comply with regulatory guidance. The closed systems are maintained filled, so that a supply of water exists on both sides of the isolation valves.

While suppression pool leakage is a function of closed system leakage for the subject penetrations, a review of Type 'C' test data for the subject CIVs showed that the valves have had low leakage rates during previous tests. This leakage is on the order of 0.6 gpm, per unit. Proposed testing of the valves under Section XI and the current requirements of the Generic Letter 8910 program will ensure valve operability.

Therefore, leakage past the CIV and out of the closed system is expected to be low and in keeping with the design basis for the suppression pool. However, the capability does exist to make-up water to the suppression pool from the Condensate Storage Tank or Spray Pond if necessary. Existing Emergency Operating Procedures require actions if suppression pool level is less than 22 feet or greater than 24 feet. Thus, the level of the suppression pool is ensured, independent of the current CIV Type 'C' testing requirement.

The proposed change to the scope of Type 'C' testing for the subject valves maintains the existing barriers to primary containment leakage, and ensures that the suppression pool level is assured for 30 days during all design basis, post-accident modes of operation. Therefore, the plant response to the design basis events is unchanged, and the proposal does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

As discussed in questions I and II, the proposed change does not alter the plant response to existing accident scenarios, and does not introduce new or different scenarios. So the margin of safety from a design basis accident standpoint is maintained.

Historically, the leakage rate through the subject valves has been determined under the Type 'C' testing program. This leakage rate has been found to be very low, and is currently on the order of 0.6 gpm. Quantifying leakage past the CIVs has been used to ensure that the suppression pool level is assured for 30 days post-accident. Under the proposed change, this leakage rate will not be quantified. This is acceptable since leakage from the suppression pool is in reality a function of closed system leakage, not solely CIV leakage. Closed system leakage is monitored and controlled by an existing Technical Specification program. Closed system leakage has been found to be very low on both units, and is currently a small fraction of a gallon per minute compared with a 5 gpm allowable. Therefore, leakage past the CIV and out of the closed system is expected to be low and in keeping with the design basis for the suppression pool. However, the capability does exist, and is proceduralized, to make-up water to the suppression pool from the Condensate Storage Tank or Spray Pond if necessary. Thus the current capability to maintain adequate suppression pool level for 30 days postaccident is assured under the proposed change.

Therefore the proposed change to the scope of Type 'C' testing for the subject valves does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: March 31, 1995.

Description of amendment request: These amendments would modify the surveillance requirement for reactor coolant system pH analysis in section 4.4.4 of the Technical Specifications (TS) for each unit. Also, they would clarify in the TS that the pH analysis would be taken at least every 72 hours whenever reactor coolant conductivity exceeds 1.0 μmho/cm.

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

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

The pH limits on reactor coolant are not affected by this change. The pH will be measured whenever it is theoretically possible for it to be outside the Tech Spec [Technical Specifications] limits of <5.6 or >8.6 (i.e., whenever the conductivity is greater than 1.0 μmho/cm). Because of the theoretical relationship between pH and conductivity as shown in Attachment A [see application dated March 31, 1995, for this reference], it is possible to establish pH limits on the reactor coolant by limiting the conductivity. As shown in this figure, the pH must be >5.6 and <8.6 if the conductivity is less than or equal to 1.0 μmho/cm. Attachment A was taken from Regulatory Guide 1.56 Revision 1, July 1978 ''Maintenance of Water Purity in Boiling Water Reactors''. As noted in both FSAR final safety analysis report and Technical Specification Bases, the pH and conductivity limits for OPERATIONAL CONDITION 1 are consistent with this theoretical relationship. The Bases for Section 3/4.4.4 of the Tech Specs [Technical Specifications] contains [contain] the following statement: ''When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits[''].

Since the conductivity is measured by grab sampling at least every 72 hours to verify that it is within limits, this will also verify that pH is within limits every 72 hours. If the conductivity should exceed 1.0 μmho/cm, pH measurements will be made to determine if the Tech Spec [Technical Specifications] pH limits have been exceeded. It should also be noted that inline conductivity instrumentation is very stable and reliable and is used to continuously monitor the reactor coolant per Tech Spec [Technical Specifications] requirements, with instrumentation connected to redundant sources (reactor water cleanup influent and reactor recirculation loop). Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

As stated above, the pH limits on reactor coolant are not affected by this change. Since the conductivity is monitored continuously, to verify that it is within limits, this will also verify that pH is within limits. If the conductivity should exceed 1.0 μmho/cm, pH measurements will be made to determine if the Tech Spec [Technical Specifications] pH limits have been exceeded. Therefore, the incorporation of this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The in-line conductivity instrumentation has been determined to be very stable and reliable in its use to continuously monitor the reactor coolant per Tech Spec [Technical Specifications] requirements. To maintain this reliability, this instrumentation is connected to redundant sources (reactor water cleanup influent and reactor recirculation loop). Based on this continuous monitoring of reactor coolant conductivity, as provided by this instrumentation, the incorporation of this change will have no impact on current safety margins, nor will it 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: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701.

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: March 31, 1995.

Description of amendment request: These amendments would delete from the Technical Specifications of each unit, the operational condition restriction in Surveillance Requirement 4.8.1.1.2.d.7 which requires that 24-hour emergency diesel generator testing be performed with at least one unit in operational condition 4 or 5 (cold shutdown or refueling).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below: The proposed changes do not:

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

The proposed change to permit the 24 hour testing of the emergency diesel generators during power operation does not increase the chances for a previously analyzed accident to occur. The function of the EDGs [emergency diesel generators] is to supply emergency power in the event of a loss of offsite power. As stated above [,] the diesel generator being tested has been determined to remain operable and available to supply the emergency loads within the required times. In addition, the three remaining EDGs will be operable during this test. Operations [Operation] of an EDGs [EDG] is not a precursor to any accident. If, however, an offsite disturbance were to occur that affected the operability of the DG [emergency diesel generator] being tested, the remaining EDGs are capable of feeding the loads necessary for safe shutdown of the plant. In summary, the proposed change does not adversely affect the performance or the ability of the diesel generators to perform their intended safety function. Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to the 24 hour surveillance requirement will not affect the operation of any safety system or alter its response to any previously evaluated accident. The diesel generator will automatically transfer from test mode if necessary to supply emergency loads in the required time. The test mode is used for the monthly surveillances of these diesel generators, resulting in no new plant operating modes being introduced. In the event the EDG fails the functional test[,] it will be declared inoperable and the actions required for an inoperable diesel will be performed. The remaining three EDGs will be operable and are capable of feeding the loads necessary for safe shutdown of the plant. Therefore, the incorporation of this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Changing the EDG test timing results is no reduction in the safety margin as defined in the design basis. Because loss of an EDG is not expected as a result of LOOP [loss of offsite power] or LOCA/LOOP [loss-of-coolant accident with a loss of offsite power] during the 24 hour test, SSES [Susquehanna Steam Electric Station] remains within its design basis. In fact, because the test EDG loads the ESS [engineered safety system] bus 8.5 seconds earlier than the non-test EDGs during LOCA with LOOP, plant response is actually improved. Risk of operation during the 24 hour EDG test is certainly less than during the current 84 hour allowed outage time (AOT) because both the impact of the initiating events evaluated (EDG in test is not actually failed) and the frequency of the limiting plant condition (loss of two EDGs) are less. No increase in frequency or impact of design basis events, and no reduction in the safety margin occurs during the 24 hour EDG test. Therefore, the incorporation of this change will have no impact on current safety margins, nor will it involve a significant reduction in the margin to [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: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701.

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

NRC Project Director: John F. Stolz.

Pennsylvania Power and Light Company, Docket No. 50–388, Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania

Date of amendment request: March 31, 1995.

Description of amendment request: This amendment would change Susquehanna Unit 2 Technical Specifications (TS) by incorporating the General Electric (GE) NRC approved methodology for GE–12 type lead use fuel assemblies (NEDE–24011–P–A–10) into the list of references in Section 6.9.3.2. The licensee plans to insert four of these fuel assemblies into the Unit 2 core during the fall of 1995. The addition of the reference to the TS would allow the use of the GE methodology to document that all applicable requirements of the safety analysis are met by the assemblies.

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

The proposed change does not:

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

Incorporation of this proposed change of adding reference NEDE–24011–P–A–10, ''General Electric Standard Application for Reactor Fuel'' to the list of references in [the] Unit 2 Technical Specifications will allow the use of the GE methodology to calculate the operating limits for the four GE Lead Use Assemblies which are of a different mechanical design than the Siemens 9X9 fuel [currently installed in the reactor core]. This NRC approved methodology will be referenced as the approved methodology in showing that all applicable safety limits of the safety analysis are met by the four GE–12 LUAs. Results of incorporating this change will not significantly increase the probability or the consequences of an accident previously evaluated.

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

As stated above, the incorporation of this change will allow the use of the GE methodology to be referenced as the approved methodology to show that all applicable limits of the safety analysis are met by the four GE–12 LUAs. Therefore, the incorporation of this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The use of the GE methodology will not result in a change in safety margin, but will ensure that the safety margin is maintained with the insertion of the four GE LUAs of the GE–12 type in Unit 2 Cycle 8. Therefore, the incorporation of these changes will have no impact on current safety margins, nor will they involve a significant reduction in the margin to [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: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701.

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: December 2, 1994.

Description of amendment request: The proposed change to Limerick Generating Station, Units 1 and 2, Technical Specifications (TS) relocates the TS Fire Protection Requirements to Licensee controlled documents consistent with NRC Generic Letter (GL) 86–10 ''Implementation of Fire Protection Requirements,'' and GL 88–12, ''Removal of Fire Protection Requirements from Technical Specifications.''

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

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

The proposed changes are administrative in nature and are consistent with NRC GL 86–10 and GL 88–12. Removal of Fire Protection Program (FPP) requirements does not affect any fire protection equipment nor plant equipment important to safety, or involve any physical modifications to plant structures, systems or components, and therefore is not associated with an accident initiator or accident mitigator and can not affect the probability of occurrence of an accident or increase the consequences of an accident. The licensee controlled Technical Requirements Manual (TRM) containing the relocated requirements will be maintained in accordance with TS Section 6.0. ''Administrative Controls'' and subject to review in accordance with 10 CFR 50.59. Since future changes to the FPP (i.e., Updated Final Safety Analysis Report and the TRM) will be evaluated per 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The changes will not alter the plant configuration (no new or different type of equipment will be installed) or create changes in methods governing normal plant operation that will introduce new failure modes. These changes will not impose different requirements and proper control of information will be maintained. These changes will not alter assumptions made in the safety analysis and licensing basis. Therefore, these changes do not create the possibility of a new or different kind of accident previously evaluated.

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

The proposed changes are administrative in nature and are consistent with NRC GL 86–10 and GL 88–12. The changes will not reduce the margin of safety since they have no impact on any safety analysis assumptions or sequence of events used in any accident analysis. In addition, any future changes to the FPP will be evaluated per the requirements of 10 CFR 50.59. Therefore, the changes do not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: January 27, 1995.

Description of amendment request: The proposed change to Limerick Generating Station (LGS) Units 1 and 2 Technical Specifications (TS) will eliminate the TS active safety function designation of eight (i.e., four per unit) Drywell Chilled Water System (DCWS) valves.

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

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

The proposed changes will eliminate the TS active safety function designation of eight (i.e., four per unit) DCWS valves. The DCWS motor operated valves (MOVs) are designated in TS as Primary Containment Isolation Valves (PCIVs), during operational conditions (OPCONS) 1, 2, and 3, which mitigate the consequences of design basis accidents. The proposed changes will prohibit the subject DCWS valves from opening during OPCONs 1, 2, and 3, thereby, eliminating the active safety function, and maintaining a passive safety function. The postulated accidents which require the Primary Containment to act as a barrier in order to mitigate the release of radioactivity described in the LGS Updated Final Safety Analysis Review [Report] (UFSAR) Section 15, are not affected by these changes. Therefore, the previously evaluated postulated on-site and off-site radiological effects of these accidents will not change.

The DCWS valves will be prohibited from opening during OPCONs 1, 2, and 3 by physical changes made to the electrical control circuitry and administrative controls. Therefore, the probability of the valves to fail in the open position will diminish, and the required Primary Containment isolation safety function will be maintained.

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

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

The proposed changes remove the affected automatic isolation relays from the DCWS MOVs' circuitry. These changes eliminate any postulated relay failure effects on the associated control circuits and electrical power supplies. The proposed changes do not introduce any new accident initiators or any new valve failure modes not previously evaluated.

Therefore, these changes will not create the possibility of a new or different kind of accident from any accidents previously evaluated.

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

The proposed changes will prohibit the opening of the DCWS valves which provide backup cooling from RECW [reactor enclosure cooling water] during OPCONs 1, 2, and 3. The RECW System is not the normal DCWS cooling alignment, is not required as a backup safety related drywell cooling system, and this backup alignment is not an automatic function. The proposed changes do not affect the function or operation of DCWS, and since the proposed changes and administrative controls ensure the valves will remain closed during OPCONs 1, 2, and 3, the capability for Primary Containment isolation is not affected. Therefore, the changes will not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: February 22, 1995.

Description of amendment request: The proposed change to Limerick Generating Station (LGS) Units 1 and 2 Technical Specifications (TS) revises various TS Surveillance Requirements to clarify the Emergency Diesel Generator acceptable steady state voltage range.

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

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

The proposed Emergency Diesel Generator steady state voltage range of 4280 [plus or minus] 120 volts provides voltages through the 4160V and 480V distribution systems which are within the operating range of the connected equipment and power system components. Therefore, the reduced steady state voltage range will not cause the malfunction of any equipment or affect the operation of any equipment in a manner which would increase the probability of occurrence of an accident previously evaluated in the [Safety Analysis Report] SAR.

Reducing the Emergency Diesel Generator steady state voltage range in the Technical Specifications maintains the capability of the Emergency Diesel Generator to start and attain rated voltage and frequency within 10 seconds and to accept the engineered safeguard loads in the required time in order to mitigate the consequences of an accident. The Emergency Diesel Generator automatic voltage regulator setting is calibrated to within a range of 4266.5 volts to 4308.5 volts. A review of results from recent monthly Emergency Diesel Generator Surveillance Tests has confirmed that the voltage regulators currently maintain the Emergency Diesel Generator steady state voltage within the 4280 [plus or minus] 120 volt range to be included in the Technical Specifications. Establishing, via Technical Specification surveillance requirements and administrative limits within Station Surveillance Test Procedures, that the Emergency Diesel Generator voltage regulator is maintaining the steady state voltage within a narrower range (within the existing range) provides increased assurance that connected equipment required to mitigate the consequences of an accident will operate as required.

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

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

Reducing the Emergency Diesel Generator steady state voltage range in the Technical Specifications to a range of 4280 [plus or minus] 120 volts does not create any new accident initiators or affect any existing accident initiators such that a different type of accident than previously evaluated could result. The function and operation of the Emergency Diesel Generators and their connected loads are not changed in a manner which would create the possibility of an accident of a different type than any previously evaluated.

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

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

Reducing the Emergency Diesel Generator steady state voltage range in the Technical Specifications to a range of 4280 [plus or minus] 120 volts does not reduce the margin of safety. The reduced range provides increased assurance that the equipment powered by the Emergency Diesel Generators will start and operate as designed in order to perform their design basis functions.

Therefore, the proposed TS changes do not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: March 1, 1995.

Description of amendment request: The proposed changes will clarify the concentrations of calibration gas required to calibrate the Hydrogen and Oxygen Analyzers, and support the requirements of Limerick Generating Station (LGS) Transient Response Implementation Plant (TRIP) T–102, ''Primary Containment Control Bases.''

Basis for proposed no significant hazards consideration determination:

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

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

The proposed TS [Technical Specification] changes remove calibration of the H2/O2 Analyzers using zero volume percent hydrogen (H2) and 100% bottled Nitrogen (N2). A calibration gas containing zero volume percent H2 and 100% bottled N2 is not required for calibration of the analyzers to the required accuracy. Calibration of the H2/O2 Analyzers is done in accordance with the manufacturer's instructions. The proposed TS changes also revise the span gas concentration from 5% to 7% to support the requirements of TRIP T–120. The H2/O2 Analyzers provide indication of the concentrations of combustible gases in the primary containment and provide annunciation when combustible gas concentrations reach unacceptable levels. Failure of the analyzers is not an accident initiator. The analyzers do not connect to the reactor coolant pressure boundary; therefore, they do not increase the probability of a LOCA [loss-of-coolant accident]. The proposed TS changes do not involve any design changes to analyzers. Therefore, these TS changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The H2/O2 Analyzers provide indication and alarms for H2 and O2 concentrations in containment. No physical or design changes to the analyzers are being made by these TS changes. During normal operations, the potential for an explosive atmosphere is negligible due to the absence of H2 sources. For Post-LOCA, conditions the levels of H2 and O2 in containment have already been evaluated in LGS UFSAR [Updated Final Safety Analysis Report] Section 6.2.5. No physical or design changes which could introduce a new analyzer failure mode are being made. The failure modes of the analyzers are evaluated in UFSAR Table 6.2–21. Therefore, these TS changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

These TS changes will clarify statements in the LGS UFSAR and TS concerning calibrated ranges of the analyzers. The change of the span gas from 5% to 7% falls within conditions previously analyzed. The Bases for TS 3/4.3.7.5 and 3/4.6.6 require operable H2/O2 Analyzers to ensure the analyzers will be available for monitoring, assessing and controlling H2 and O2 in containment following a LOCA. These TS changes do not adversely affect operability of the analyzers or their availability for use during Post-LOCA conditions; therefore, the margin of safety is not reduced.

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

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

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

NRC Project Director: John F. Stolz.

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

Date of amendment request: December 15, 1994.

Description of amendment request: In accordance with 10CFR50.90, PSE&G proposes to remove Technical Specification Surveillance Requirement 4.8.1.1.2.h.1, and utilize plant-controlled programs to govern diesel generator maintenance. To ensure procedural consistency and reduce the impact of this change on Hope Creek procedures, the remaining Surveillance Requirements of Technical Specification 4.8.1.1.2.h are not renumbered.

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

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

The proposed change is consistent with the improved Standard Technical Specifications (NUREG–1433) and does not result in any changes to the existing plant design. The Hope Creek preventative maintenance program will utilize diesel generator performance history, engineering analyses and manufacturer's recommendations as appropriate for determining diesel generator inspection requirements. Since the changes do not impact the ability of the diesel generators and the AC electrical power sources to perform their function, the changes do not result in a significant increase in the consequences of any accident previously evaluated. The diesel generators will continue to function as designed. Therefore, the proposed change will not impact the probability of occurrence of any accident previously evaluated.

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

This request does not result in any change to the plant design nor does it involve a significant change in current plant operation. The diesel generators will be inspected utilizing diesel generator operating history, engineering analyses and manufacturer's recommendations as appropriate, and the remaining surveillance requirements will not be changed. As a result, no new failure modes will be introduced, and the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will not involve a significant reduction in a margin of safety. The proposed request does not adversely impact the reliability of the diesel generators. As stated above, the diesel generator operating history, engineering analyses and the manufacturer's recommendations will be utilized as appropriate to perform the diesel generator inspections. In addition, the diesel generators will continue to perform their design functions. This request does not involve an adverse impact on diesel generator operation or reliability. Since the diesel generator function is not affected by the proposed changes, this request does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070.

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

NRC Project Director: John F. Stolz.

Tennessee Valley Authority, Docket No. 50–327, Sequoyah Nuclear Plant, Unit 1, Hamilton County, Tennessee

Date of amendment request: April 6, 1995 (TS 95–09).

Description of amendment request: The proposed change would revise Operating Condition 2.C.(25) to extend the ice condenser Surveillance 4.6.5.1.d to October 1, 1995, to coincide with the Unit 1 Cycle 7 refueling outage.

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

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

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

The proposed change is temporary and allows a one-time extension of the ice condenser Surveillance Requirement 4.6.5.1.d for Cycle 7 to allow surveillance testing to coincide with the seventh refueling outage. The proposed surveillance interval extension will not cause a significant reduction in system reliability nor affect the ability of the system to perform the design function. Current monitoring of plant conditions and continuation of the surveillance testing required during normal plant operation will continue to be performed to ensure conformance with TS operability requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Extending the surveillance interval for the performance of ice condenser testing will not create the possibility of a new or different kind of accident. No change is required to any system configuration, plant equipment, or analyses.

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

The safety limits assumed in the accident analyses and the design function of the equipment required to mitigate the consequences of postulated accidents will not be changed significantly. Existing analysis indicates that the potential reduction in ice weight resulting from the proposed extension will continue to maintain the maximum containment accident pressure below 12 pounds per square inch gauge. The ice condenser will continue to support accident mitigation functions although some Row 1 baskets could drop slightly below the required 993-pound analysis limit. Therefore, the plant will be maintained with acceptable ice weights for accident mitigation and the proposed extension will not significantly reduce the margin of safety.

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

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

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: April 6, 1995 (TS 95–11).

Description of amendment request: The proposed change would relocate the constant numerical value found in the overtemperature delta temperature and overpower delta temperature equations of Technical Specifications Table 2.2–1 and place them in the Core Operating Limits Report (COLR). This would be accomplished by revising notes 1 and 2 in Table 2.2–1 to state that the values are located in the COLR. The values of the constants, however, would not be changed. Also, the ''Overtemperature and Overpower Delta Temperature Setpoint Parameter Values for Specification 2.2.1'' would be added to the list of core operating limits specified in Section 6.9.1.14 that are required to be included in the COLR. In addition, a reference to WCAP–8745–P–A, ''Design Bases for the Thermal Overpower delta-T and Thermal Overtemperature delta-T Trip Functions,'' would be added to Section 6.9.1.14.a.

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

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

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

The proposed changes will allow changes to the constant numerical values for the overtemperature delta temperature (OT[delta-]T) and overpower delta temperature (OP[delta-]T) equations in accordance with the 10 CFR 50.59 requirements. This revision does not revise these constants, but relocates them to the core operating limits report (COLR) that is governed by the 10 CFR 50.59 requirements. The addition of the lag compensator functions for measured [delta-]T and average temperature in these equations does not alter the setpoint because this lag function has a value of unity. Therefore, the proposed revision does not alter plant functions or setpoints, but does allow for a more timely revision process for parameters that may require changes due to specific fuel cycle requirements or updated plant analyses. The use of the lag functions and revisions to the constant numerical values will be maintained within the safety analysis for the plant by the 10 CFR 50.59 process requirements. The probability of an accident is not increased because the plant functions are not altered by the proposed revision and future changes will be in accordance with 10 CFR 50.59. Additionally, the consequences of an accident are not increased because the mitigation functions of the OT[delta-]T and OP[delta-]T functions are not changed and revisions to the equations that derive the setpoints will be processed under the requirements of the 10 CFR 50.59 program.

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

The proposed revision will not change plant functions and future revisions will continue to be controlled in accordance with the 10 CFR 50.59 requirements. The addition of the lag functions does not create a new accident potential because these functions have already been considered in the analysis as shown in NUREG 1431. Therefore, the possibility of a new or different kind of accident is not created by the proposed revision.

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

Plant parameters are not altered by the proposed revision and the OT[delta-]T and OP[delta-]T functions will not reflect a change in setpoint generation or value. The proposed change will allow revision of the constant numerical values and use of the lag compensator functions in accordance with the 10 CFR 50.59 provisions to ensure the design basis of the plant is maintained. This revision does not result in changes that reduce the margin of safety because the OT[delta-]T and OP[delta-]T functions remain unchanged and future revisions to these functions will be performed in accordance with 10 CFR 50.59.

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

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

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: April 6, 1995 (TS 94–15).

Description of amendment request: The proposed change would modify the Technical Specifications associated with the Post Accident Sampling (PAS) system by deleting License Condition 2.C.(23)F for Unit 1 and 2.C.(16)g for Unit 2 that require operation of the PAS system in accordance with referenced letters no later than startup from the second refueling outage. The submittal also includes a revised description of operation of the PAS system for insertion into the Updated Final Safety

Analysis Report for staff approval. This information supersedes the information contained in the letters referenced in the License Conditions listed above and would be maintained in accordance with the 10 CFR 50.59 process.

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

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

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

The proposed change involves the deletion of license conditions that authorized TVA to operate SQN's postaccident sampling (PAS) system. TVA proposed change establishes programmatic control of SQN's PAS system under SQN TS 6.8.4.e and the SQN Final Safety Analysis Report. Any future changes to SQN's PAS Program would be governed by the 10 CFR 50.59 process. PAS and analysis will continue at SQN through grab sample acquisition and laboratory analysis and will continue to meet the PAS objectives in NUREG–0737, Item II.B.3 and Regulatory Guide 1.97, Revisions 2. Accordingly, the proposed change does not affect the probability or consequences of an accident.

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

The proposed change involves improvements in the operational reliability of SQN's PAS system by using more reliable laboratory analysis methods, reducing sampling personnel radiation dose, and incorporating practical methods for sample acquisition and analysis. Because the proposed change involves license conditions and sampling methods that are utilized for postaccident recovery, the potential for an unanalyzed accident is not created. Consequently, no new failure modes are introduced.

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

Plant safety margins are established through limiting conditions of operation, limiting safety system settings, and safety limits specified in the TSs. As a result of the proposed amendment, there will be no change to either the physical design of the plant or to any of these settings and limits. The proposed changes do not affect the safe operation of SQN. Therefore, there are no changes to any of the margins of safety.

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

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

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: April 6, 1995 (TS 94–18).

Description of amendment request: The proposed change would revise Surveillance Requirement (SR) 4.0.5 by replacing the current Inservice Inspection program requirements with the requirements stated in the Standard Technical Specifications (NUREG–1431). As a result, SR 4.0.5 would more clearly specify the inservice inspection (ISI) program requirements and the inservice testing (IST) program requirements of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components. The licensee also proposed deletion of Technical Specification 3/4.4.10, ''Structural Integrity ASME Code Class 1, 2 and 3 Components,'' and its related Bases information.

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

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

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

Operation of the facility in accordance with the proposed amendment would not involve any increase in the probability of occurrence or consequences of an accident previously evaluated. The Inservice Inspection and Testing Programs, pursuant to 10 CFR 50.55a are described in the TSs. The proposed amendment, in accordance with NUREG–1431 and draft NUREG–1482 permits relief from an American Society of Mechanical Engineers (ASME) code requirement in the interim between the time of submittal of a relief request and NRC approval of the relief. The changes being proposed do not affect assumptions contained in plant safety analyses or change the physical design and/or operation of the plant, nor do they affect TSs that preserve safety analysis assumptions. Any relief from the approved ASME Section XI code requirements that is implemented prior to NRC review and approval will require evaluation under the 10 CFR 50.59 process to determine that no TS changes or unreviewed safety questions exist. This evaluation process will ensure that the impact of any code relief is thoroughly evaluated and that the structures, systems, and components remain in conformance with assumptions made in the safety analysis. The proposed change to delete SQN TS 3/4.4.10, Structural Integrity, does not affect plant safety analyses or change the physical design or operation of the plant. The proposed amendment relocates the structural integrity requirements under the existing TS Surveillance Requirement (SR) 4.0.5 to allow these requirements to be governed and controlled within the inservice inspection (ISI) program. Therefore, operation of the facility in accordance with the proposed amendment would not affect the probability or consequences of an accident previously evaluated.

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

Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. The Inservice Inspection and Testing Programs, pursuant to 10 CFR 50.55a are described in the TSs. The proposed amendment, in accordance with NUREG–1433 and draft NUREG–1482, permits relief from an ASME code requirement in the interim between the time of submittal of a relief request and NRC approval of the relief. The changes being proposed will not change the physical plant or the modes of operation defined in the Facility License. The changes do not involve the addition or modification of equipment nor do they alter the design or operation of plant systems. Any relief from the approved ASME Section XI code requirements that is implemented prior to NRC review and approval will require evaluation under the 10 CFR 50.59 process to determine that no TS changes or unreviewed safety questions exist. This evaluation process will ensure that the impact of the code relief is thoroughly evaluated and that the structures, systems, and components remain in conformance with assumptions made in the safety analysis. The proposed change to delete SQN TS 3/4.4.10 does not affect plant safety analyses or change the physical design or operation of the plant. The proposed amendment relocates the structural integrity requirements under the existing TS SR 4.0.5 to allow these requirements to be governed and controlled within the ISI program. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Operation of the facility in accordance with the proposed amendment would not involve any reduction in a margin of safety.

The Inservice Inspection and Testing Programs, pursuant to 10 CFR 50.55a, are required by the SQN TSs. The proposed amendment, in accordance with NUREG–1431 and draft NUREG–1482 permits relief from an ASME code requirement in the interim between the time of submittal of a relief request and NRC approval of the relief. Any relief from the ASME Section XI code is required to be evaluated under the 10 CFR 50.59 process to determine that no TS changes or unreviewed safety questions exist. This evaluation process will ensure that code relief does not affect the ability of structures, systems, or components to perform their design function, affect compliance with any TS requirements or reduce the margin of safety. The proposed change to delete SQN TS 3/4.4.10 does not affect plant safety analyses or change the physical design or operation of the plant. The proposed amendment relocates the structural integrity requirements under the existing TS SR 4.0.5 to allow these requirements to be governed and controlled within SQN's ISI program. Therefore, operation of the facility in accordance with the proposed amendment would not involve a reduction in the margin of safety.

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

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

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: April 6, 1995 (TS 95–06).

Description of amendment request: The proposed change would delete the technical specification requirement that limits and controls loads traveling over the spent fuel pool (Specification 3.9.7), the graph that relates the Load Carried Over the Shield to the Allowable Height Above the Shield Surface (Figure 3.9–1), the crane interlocks and physical stops surveillance requirements (Specifications 4.9.7.1 and 4.9.7.2), and the related Bases information. These requirements would be relocated to administratively controlled procedures.

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

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

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

The proposed TS change involves the relocation of a requirement that does not pertain to limitations or conditions of reactor operation or to equipment to mitigate design basis accidents or transients. SQN is proposing to relocate this TS based on NRC's final policy statement on TS improvement (58 FR 39132, dated July 22, 1993). Based on this criteria, the spent fuel pit (SFP) crane travel is not important to operational safety and may be relocated to administratively controlled procedures. By placing the crane travel requirements in administratively controlled procedures, adequate controls will remain in place to prevent heavy loads from traveling over fuel assemblies in the SFP. The administratively controlled procedure that controls the by-passing of the interlocks and physical stops is subject to the requirements of TS 6.5.1A. Therefore, the relocation of this TS will not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed change involves relocating TS requirements to another administratively controlled document. No modifications to the plant are involved. Additionally, there are no changes to the operation of the plant or equipment proposed. Based on this, the relocation of this TS will not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed change involves the relocation of TS requirements to administratively controlled procedures. The relocation of this requirement is based on the criteria endorsed in the Commission's Policy Statement on TS improvements as it pertains to 10 CFR 50.36. Additionally, this change does not alter the basic design and safety analysis requirements, as discussed in the Updated Final Safety Analysis Report. Therefore, the deletion of this TS will not involve a reduction in the margin of safety.

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

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

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: April 6, 1995 (TS 94–19).

Description of amendment request: The proposed change would revise the Action statement for Technical Specification 3.8.1.1 by inserting a new Action a, relabeling and modifying existing Action a to become Action b, adding a footnote referenced to Action b, renumbering the subsequent action statements, and adding information to the Bases that amplifies the action statements. The proposed new Action a would no longer address required actions for diesel generator testing. It would require that, should one of the AC electrical power sources listed be inoperable, then operability of the remaining offsite AC circuit be demonstrated by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If two offsite circuits cannot be restored within 72 hours, the unit must be placed in hot standby within the next 6 hours and in cold shutdown within the following 30 hours.

The proposed change to Action b would address the testing requirements should a diesel generator become inoperable. It would require testing of operable diesel generators if the inoperability of the affected diesel generator has the potential to be the result of a common cause failure. A footnote would clarify that the common cause determination must be completed regardless of how long the diesel generator inoperability persists or Surveillance 4.8.1.1.2.a.4 must be completed to verify diesel generator operability. The proposed change to the Bases would provide clear guidance on the use of common cause failure determinations to eliminate unnecessary diesel generator testing and would define when testing is required to verify diesel generator operability.

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

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

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

The proposed revisions do not alter the plant features or operating practices. This revision will reduce unnecessary starts of the diesel generators (D/Gs) when a common cause failure is not involved or for an inoperable offsite circuit. This change will not affect the accident mitigation capabilities of the D/Gs, but should improve the reliability by reducing the wear and tear associated with starting the D/Gs. The D/Gs are not the source of a postulated accident and because this change does not alter plant functions or operating practices the probability of an accident is not increased. The D/G's operability will continue to be verified for conditions that indicate a potential common-cause failure to ensure accident mitigation capabilities are not affected. Therefore, this revision will continue to provide actions that will support alternating-current (ac) electrical power source safety functions without unnecessary degradation of the D/Gs and will not increase the consequences of an accident.

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

The D/Gs are not the source of accidents and the proposed revision will not alter plant functions or actions by more appropriately limiting the conditions when a D/G must be verified operable. Therefore, the possibility of a new or different accident is not created

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

This revision does not alter plant functions that provide the margin of safety. The reduction of D/G testing will only be allowed for situations where the operable D/Gs are not affected by the conditions resulting in the ac power source inoperability. This reduced testing should improve D/G reliability for accident mitigation functions and further ensure the margin of safety provided by the D/Gs. Therefore, the margin of safety is not reduced by the proposed revision to limit unnecessary D/G starts.

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

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

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: April 6, 1995 (TS 95–02).

Description of amendment request: The proposed change would add Limiting Condition for Operation 3.0.6 to allow equipment that has been removed from service or declared inoperable to be returned to service under administrative control in order to perform testing required to demonstrate operability. It would be applicable for operability testing of the inoperable equipment or other equipment that requires the operability feature to be in service in order to perform the test. A related change to the Bases would provide amplifying explanation on the use of this new provision. In addition, a proposed change to Action 18 of Table 3.3–3, ''Engineered Safety Feature Actuation System Instrumentation,'' would clarify the time interval that an instrument channel may be in the bypass condition. For those instruments that reference Action 18, the change would allow the bypass for 6 hours.

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

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

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

The addition of the exception to TSs 3.0.1 and 3.0.2 and the definition for the time to place a channel in bypass will not change plant equipment or the operating practices at SQN. The exception will allow testing to be performed with inoperable equipment returned to service under administrative controls, but will not change functions. The function will be available from other redundant channels during the brief durations that the new provision would be utilized. The specified time interval to achieve a bypass condition will clarify the implementation of the action requirement with the affected functions remaining available through the redundant operable channels. This clarification does not change the intent of the action but does set the previously undefined time interval.

The proposed change affects actions associated with the actuation of functions to mitigate accidents and are not the source of an accident. Therefore, the probability of an accident is not increased. The affected functions provide accident mitigation functions and the proposed revisions serve to ensure equipment can be maintained in required conditions within acceptable time intervals and administrative controls. The brief periods utilized for the TSs 3.0.1 and 3.0.2 exception will not significantly affect the accident mitigation capabilities because of the availability of redundant equipment. In addition, the benefit of performing operability testing to return equipment permanently to service or to maintain the operability of other equipment outweighs the slight reduction in safety function actuation redundancy. Therefore, the proposed change will not significantly increase the consequences of an accident.

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

The proposed changes affect functions utilized to mitigate an accident and are not the source of an accident. The exception provides reasonable flexibility to maintain equipment operability and the bypass time interval reduces the potential for damage to safety related equipment. Because plant functions are not changed as a result of this request the possibility of a new or different kind of accident is not created.

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

The proposed change does not alter setpoints or operating considerations that maintain the margin of safety for SQN. These changes provide needed flexibility to perform TS required testing and clarifications for implementing action requirements. These changes will slightly affect the redundancy of the affected safety functions but provide greater benefit for maintaining equipment in an operable condition. Therefore, the margin of safety provided by the affected equipment has not changed and the proposed change will not result in a reduction.

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

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

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: April 6, 1995 (TS 94–04).

Description of amendment request: The proposed amendment would change the power range neutron flux channel calibration frequency surveillance requirement from monthly to every 31 effective full power days and delay the requirement to perform the surveillance for 96 hours after reaching 15 percent power. A proposed change to the Bases would provide amplifying information.

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

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

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

The likelihood that an accident will occur is neither increased or decreased by this TS change, which only affects when the first surveillance is performed following an outage and changes the frequency of performance of the surveillance. Before start-up following refueling outage, the power range high trip setpoint is set below 85 percent power, typically 60 percent, for conservatism. The power range low trip setpoint is set at 22 percent power, TS requires the setpoint to be less than or equal to 25 percent power. These settings are in addition to the conservatism built into start-up following a refueling outage. Therefore, delaying the first performance for 96 hours will not impact on the operation of the plant since the setpoints are set conservatively. Also, the change of the frequency to every 31 effective full power days (EFPD) only delays the surveillance when the plant is operated at reduced power. During operation at reduced power changes in the neutron flux are also reduced. Therefore, changing the frequency from monthly to every 31 EFPD allows slow changes in neutron flux during the fuel cycle to be more accurately detected and evaluated.

This TS change will not impact the function or method of operation of plant equipment. Thus, there is not a significant increase in the probability of a previously analyzed accident due to this change. No systems, equipment, or components are affected by the proposed change. Thus, the consequences of a malfunction of equipment important to safety previously evaluated in the Updated Final Safety Analysis Report are not increased by this change.

The proposed changes provide TS improvements that ensure the system operates within the bounds of SQN's accident analysis as contained in the Final Safety Analysis Report (FSAR) and only affects when a surveillance is performed. This change has no impact on accident initiators and does not involve a physical modification to the plant. Accordingly, the proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

This revision will not change any plant equipment, system configurations, or accident assumptions. This change will more accurately monitor changes in the condition of the core.

Fuel burn-up is necessary to change the relationship between the incore axial power and the excore detectors response. At reduced levels the effectiveness of the monitoring activity is reduced. Therefore, changing the frequency to 31 EFPD allows slow changes in neutron flux during the fuel cycle to be more accurately detected and evaluated. Delaying the first performance of the surveillance requirement, until 96 hours after reaching 15 percent rated thermal power, will allow the unit to be in a more stable condition. Therefore, this change will not affect the safety function of any components and will create the possibility of a new or different kind of accident.

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

The proposed changes provide TS improvements for SQN's power range monitoring system that ensure the system operates within the bounds of SQN's accident analysis as contained in the FSAR since only the time interval between performances of the surveillance is being extended. This change does not involve a physical modification to SQN's power range monitoring system. Accordingly, the margin of safety has not been reduced.

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

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

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

NRC Project Director: Frederick J. Hebdon.

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

Date of amendment request: April 6, 1995 (TS 95–08).

Description of amendment request: The proposed change would (1) change the core alteration definition to limit the term to reactor vessel internal activities that could have an affect on core reactivity, (2) change the quadrant power tilt ratio definition to eliminate the conflict in the definition of the term and its use in Surveillance Requirement 4.2.4.2, and (3) revise the Unit 1 Operational Modes parameters in Table 1.1 to be consistent with the description in Table 1.1 for Unit 2.

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

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

1. Involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes provide TS improvements that ensure the plant operates within the bounds of SQN's accident analysis as contained in the Final Safety Analysis Report (FSAR) and only affects the definitions and does not have any affect on any work performed. The change to core alteration is to clarify those components that may result in reactivity changes. The change will not effect movement of fuel or components that effect reactivity, therefore, a fuel handling accident will not be effected. The change in the definition of quadrant power tilt ratio (QPTR) allows the alternate method of determining QPTR to be utilized. The current TS surveillance requirement (SR) and bases allow alternate means for determining QPTR, therefore, revising the definition will have no effect on any accident. The revision to the mode parameters is administrative in nature, therefore it will have no effect on any accident. This change has no impact on accident initiators and does not involve a physical modification to the plant. Accordingly, the proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

This revision will not change any plant equipment, system configurations, or accident assumptions. This change will better define the associated parameters and will eliminate potential ambiguity and confusion. The change in the definition of core alteration allows components that do not affect reactivity to be moved within the reactor vessel. The change in the definition will not effect the monitoring of QPTR with one channel inoperable. The core will be monitored in accordance with the SRs. Therefore, this change will not affect the safety function of any components and will not create the possibility of a new or different kind of accident.

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

The proposed changes provide improvements for SQN's TS. This change does not involve a physical modification to the plant nor change the methods of monitoring plant parameters. Accordingly, the margin of safety has not been reduced.

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

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

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

NRC Project Director: Frederick J. Hebdon.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: March 24, 1995.

Description of amendment request: This amendment request proposes to revise Technical Specification (TS) 1.7, ''Containment Integrity,'' TS 3/4.6.1, ''Containment Integrity,'' TS 3/4.6.3, ''Containment Isolation Valves,'' and the associated Bases. These proposed changes will relocate TS Table 3.6–1, ''Containment Isolation Valves,'' to Wolf Creek Generating Station procedures. This proposed change is in accordance with the guidance provided in Generic Letter 91–08, ''Removal of Component Lists from Technical Specifications,'' dated May 6, 1991.

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

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

The proposed changes simplify the technical specifications, meet the regulatory requirements for control of containment isolation, and are consistent with the guidelines of GL 91–08. The procedural details of Technical Specification Table 3.6–1 have not been changed, but only relocated to a different controlling document. The proposed changes are administrative in nature, should result in improved administrative practices, and do not affect plant operations.

The probability of occurrence of a previously evaluated accident is not increased because this change does not introduce any new potential accident initiating conditions. The consequences of an accident previously evaluated is not increased because the ability of [the] containment to restrict the release of any fission product radioactivity to the environment will not be degraded by this change.

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

The proposed changes are administrative in nature, do not result in physical alterations or changes to the operation of the plant, and cause no change in the method by which any safety-related system performs its function. Therefore, this proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The administrative change to relocate Technical Specification Table 3.6–1 to appropriate plant procedures does not alter the basic regulatory requirements for containment isolation and will not adversely affect containment isolation capability for credible accident scenarios. Adequate control of the content of the table is assured by existing plant procedures.

The proposed relocation of Technical Specification Table 3.6–1 does not alter the requirements for containment isolation valve operability currently in the technical specifications. The LCO [limiting condition for operation] and Surveillance Requirements would be retained in the revised technical specifications. Therefore, the proposed change will not affect the meaning, application, and function of the current technical specification requirements for the valves in Table 3.6–1.

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 locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621.

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Project Director: William H. Bateman.

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.

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

Date of amendment request: April 4, 1995, as supplemented by letter dated April 5, 1995.

Description of amendment request: The proposed amendment would change the technical specifications on moderator temperature coefficient. The proposed change constitutes a one time deviation not to perform the two-thirds end-of-cycle moderator temperature coefficient test for Cycle 7.

Date of individual notice in the **Federal Register**: April 11, 1995 (60 FR 18432).

Expiration date of individual notice: May 11, 1995.

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

Notice of Issuance of Amendments to Facility Operating Licenses

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

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the **Federal Register** as indicated.

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

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

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

Date of application for amendment: February 24, 1995.

Brief description of amendment: The amendment revises Technical Specification Section 4.6.1.2.a, Primary Containment/Containment Leakage, to reference 10 CFR Part 50, Appendix J, as modified by approved exemptions.

Date of issuance: April 10, 1995.

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

Amendment No.: 183.

Facility Operating License No. DPR–69: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 8, 1995 (60 FR 12789).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 10, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: September 30, 1994, as supplemented on March 24, 1995.

Brief Description of amendments: The proposed change will revise Technical Specification requirements to eliminate the reactor scram and isolation functions of the Main Steam Line Radiation Monitors. The March 24, 1995, supplement provided clarifying information only and did not affect the NRC's determination of no significant hazards considerations.

Date of issuance: March 31, 1995.

Effective date: March 31, 1995.

Amendment No.: 160.

Facility Operating License Nos. DPR–71 and DPR–62. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 9, 1994 (59 FR 55867). The March 24, 1995, submittal provided clarifying information only and did not affect the no significant hazards consideration as published in the **Federal Register**.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 31, 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.

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of application for amendment: July 22, 1994, as supplemented March 6, 1995.

Brief description of amendment: The amendments change the Technical Specifications to implement a performance based assessment program, including corresponding organizational and functional changes. Specifically, the changes affect the independent assessment of plant activity and the independent review function, the independent assessment of plant activity and the Independent Safety Engineering Group. These functions will be performed by the Nuclear Assessment Section (NAS). The NAS's fundamental role will be to: (1) Assist plant management in the early identification of issues that may prevent the plant from achieving quality, and (2) ensure effective correction of deficiencies.

Date of issuance: March 31, 1995.

Effective date: March 31, 1995.

Amendment No.: 160.

Facility Operating License No. DPR–23. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: August 31, 1994 (59 FR 45017). The March 6, 1995, submittal added Radiation Protection to the list of assessments in TS 6.5.5.2 and reworded Section 6.5.4.4, but did not change the no significant hazards consideration determination as published in the **Federal Register**.

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

No significant hazards consideration comments received: No.

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550.

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of application for amendment: August 23, 1994, as supplemented March 2, 1995.

Brief description of amendment: The amendment increases the trip voltage settings of the degraded grid voltage relays which are shown in TS Table 3.5–1, Engineering Safety Feature System Initiation Instrument Setting Limits, Item 6b.

Date of issuance: April 14, 1995.

Effective date: April 14, 1995.

Amendment No.: 161.

Facility Operating License No. DPR–23. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: March 2, 1995 (60 FR 11692). The licensee's March 2, 1995, submittal provided clarifying information that did not affect the no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 14, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550.

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

Date of application for amendments: June 9, 1994.

Brief description of amendments: The amendments implement a partial application of the Generic Electric ARTS (Average Power Range Monitor (APRM)/Rod Block Monitor (RBM)/Technical Specification) Improvement Program. Four new ARTS thermal limits replace the existing flow-referenced APRM trip setpoint setdown requirements and the Minimum Critical Power Ratio (MCPR) K_f factor.

Date of issuance: April 13, 1995.

Effective date: Immediately to be implemented within 60 days.

Amendment Nos.: 103 and 88.

Facility Operating License Nos. NPF–11 and NPF–18. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 3, 1994 (59 FR 39582).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 13, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 29, 1993, as supplemented October 8, 1993.

Brief description of amendment: The amendment revises the Technical Specification ACTION STATEMENTS related to operability of the control room emergency filtration system. A portion of the amendment request was denied. A separate Notice of Denial of Amendment has been sent to the **Federal Register** for publication.

Date of issuance: March 31, 1995.

Effective date: March 31, 1995, with full implementation within 45 days.

Amendment No.: 103.

Facility Operating License No. NPF–43. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: August 18, 1993 (58 FR 43925). The October 8, 1993, letter provided clarifying information within the scope of the original submittal and did not change the staff's initial no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: January 18, 1995.

Brief description of amendments: The amendments revise TS Table 4.3–3 to allow the analog channel operational test interval for radiation monitoring instrumentation to be increased from monthly to quarterly, and are consistent with the guidance in Generic Letter 93–05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation.''

Date of issuance: April 3, 1995.

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

Amendment Nos.: 154 and 136.

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

Date of initial notice in **Federal Register**: March 1, 1995 (60 FR 11132).

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 22, 1993.

Brief description of amendment: The amendment revised operability requirements for the Reactor Protection System and the Engineered Safety Features Acturation System.

Date of issuance: April 3, 1995.

Effective date: April 3, 1995.

Amendment No.: 159.

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

Date of initial notice in **Federal Register**: September 1, 1993 (58 FR 46229).

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 19, 1994, resubmitted October 20, 1994, and supplemented February 20, 1995.

Brief description of amendments: These amendments relate to the maximum allowable reactor thermal power operation with inoperable main steam safety valves.

Date of issuance: April 11, 1995.

Effective date: April 11, 1995.

Amendment Nos.: 172 and 166.

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

Date of initial notice in **Federal Register**: November 23, 1994 (59 FR 60380).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 11, 1995. The February 20, 1995 submittal provided additional information that did not change the staff's proposed no significant hazards consideration determination.

No significant hazards consideration comments received: No.

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

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: October 28, 1994.

Brief description of amendment: The amendment revised the license by deleting the ''Plan for the Integrated Scheduling of Plant Modifications for the Duane Arnold Energy Center,'' as a condition of the license.

Date of issuance: April 3, 1995.

Effective date: April 3, 1995.

Amendment No.: 208.

Facility Operating License No. DPR–49. Amendment revised the license.

Date of initial notice in **Federal Register**: March 1, 1995 (60 FR 11134).

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: December 6, 1994, as supplemented on February 27, and April 4, 1995.

Brief description of amendment: The amendment revises the Technical Specifications to allow the use of the Combustion Engineering sleeving process for repairing steam generator tubes. (The current requirement specifies that degraded steam generator tubes be repaired by plugging.)

Date of issuance: April 14, 1995.

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

Amendment No.: 149.

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

Date of initial notice in **Federal Register**: January 18, 1995 (60 FR 3673). The February 27, and April 4, 1995, letters 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 April 14, 1995.

No significant hazards consideration comments received: Yes.

Comments were provided by letter dated February 17, 1995, from the Maine State Nuclear Safety Inspector, Office of Nuclear Safety, Division of Health Engineering, Department of Human Services. The NRC staff responded to his comments in its letter dated March 15, 1995.

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

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: September 30, 1994, as supplemented February 13, 1995.

Brief description of amendment: The amendment revises the Technical Specifications to (1) clarify the definition of core alterations, (2) change the verbiage in the Limiting Condition For Operation (LCO) addressing the refueling operations, (3) make changes of surveillance requirements involving source range instrumentation, and (4) change the LCO regarding the Residual Heat Removal and coolant circulation water levels to be consistent with the guidance provided in NUREG–1431, Standard Technical Specifications for Westinghouse plants.

Date of issuance: April 12, 1995.

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

Amendment No.: 107.

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

Date of initial notice in **Federal Register***:* November 9, 1994 (59 FR 55877). The February 13, 1995, 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 April 12, 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.

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: January 26, 1993, as supplemented August 4, 1993.

Brief description of amendment: The amendment changes the Technical Specifications governing electrical power systems, AC and DC power sources, and onsite power distribution for shutdown conditions (modes 5 and 6).

Date of issuance: April 12, 1995.

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

Amendment No.: 108.

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

Date of initial notice in **Federal Register:** May 12, 1993 (58 FR 28058). The August 4, 1993, submittal 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 April 12, 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.

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

Date of amendment request: April 23, 1993.

Description of amendment request: The amendment revises the Appendix A Technical Specifications to allow longer surveillance test intervals and allowed outage times for the reactor protection system and the engineered safety features actuation system. Also, the amendment removes the requirement to perform the reactor trip system analog channel operational test on a staggered basis.

Date of issuance: April 10, 1995.

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

Amendment No.: 36.

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

Date of initial notice in **Federal Register**: August 4, 1993 (58 FR 41507).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 10, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 16, 1994 (Reference LAR 94–03)

Brief description of amendments: The amendments revise TS 4.6.1.2, ''Containment Integrity,'' to allow a more flexible schedule for testing the primary containment integrated leakage rate.

Date of issuance: April 11, 1995.

Effective date: April 11, 1995, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 1—Amendment No. 99; Unit 2—Amendment No. 98

Facility Operating License Nos. DPR–80 and DPR–82. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 30, 1994 (59 FR 14893).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 11, 1995.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: August 17, 1994.

Brief description of amendments: The amendments revise the combined technical specifications (TS) to change TS 3/4.4.9.1, ''Reactor Coolant System—Pressure/Temperature Limits,'' Figures 3.4–2, ''Reactor Coolant System Heatup Limitations—Applicable Up to 8 EFPY,'' and 3.4–3, ''Reactor Coolant System Cooldown Limitations—Applicable Up to 8 EFPY,'' to extend the applicability up to 12 effective full-power years (EFPYs). TS 3/4/4/9/3, ''Overpressure Protection Systems,'' is revised to specify a new low-temperature overprotection (LTOP) system actuation pressure setpoint. The associated Bases were also appropriately revised. Additionally, TS 3/4.1.2.2, ''Flow Paths—Operating;'' TS 3/4.1.2.4, ''Charging Pumps—Operating;'' TS 3/4.4.1.3, ''Hot Shutdown;'' TS 3/4.4.1.4.1, ''Cold Shutdown—Loops Filled;'' TS 3/4.4.9.3, ''Overpressure Protection Systems;'' and TS 3/4.5.3, ''ECCS Subsystems—T_{avg} Less than 350 Degrees F,'' are revised to specify a new LTOP system enable temperature.

Date of issuance: April 13, 1995.

Effective date: April 13, 1995.

Amendment Nos.: Unit 1—Amendment No. 100; Unit 2—Amendment No. 99.

Facility Operating License Nos. DPR–80 and DPR–82. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 12, 1994 (59 FR 51622).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 13, 1995.

No significant hazards consideration comments received: No.

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

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: June 13, 1994.

Brief description of amendment: The amendment removes License Condition 2.E from the Facility Operating License. License Condition 2.E incorporated the requirements of U.S. Department of Interior publication ''Environmental Criteria for Electric Transmission Systems''—1970, which applies to the construction cleanup, restoration, and maintenance of transmission lines. The NRC staff has determined that removing this condition from the Facility Operating License has no bearing on plant safety or the health and safety of the public, and is therefore acceptable.

Date of issuance: March 31, 1995.

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

Amendment No.: 224.

Facility Operating License No. DPR–59. Amendment revised the Facility Operating License.

Date of initial notice in **Federal Register**: March 1, 1995 (60 FR 11140).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 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.

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: June 10, 1995.

Brief Description of amendments: The amendments allow modifications to be made for both units to relocate the lower level steam generator water level taps during the upcoming refueling outages. These modifications affect the Technical Specifications associated with the reactor trip system and engineered safety feature actuation system setpoints.

Date of issuance: April 7, 1995.

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

Amendment Nos.: 114 and 105.

Facility Operating License Nos. NPF–2 and NPF–8. Amendments revise the Technical Specifications.

Date of initial notice in **Federal Register**: March 6, 1995 (60 FR 12253).

Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 7, 1995.

No significant hazards consideration comments received: No.

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

Southern Nuclear Operating Company, Inc., Joseph M. Farley Nuclear Plant, Unit 2, Houston County, Alabama

Date of application for amendment: December 7, 1994, as supplemented February 14 and March 20, 1995.

Brief description of amendment: The December 7, 1994, submittal requested a permanent change to the Technical Specifications for both units related to steam generator tube support plate voltage-based repair criteria in accordance with the draft Generic Letter on this issue.

Date of issuance: April 7, 1995.

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

Amendment No.: 106.

Facility Operating License No. NPF–8. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: February 15, 1995 (60 FR 8754). The February 14 and March 20, 1995, letters provided clarifying information that did not change the scope of the original December 7, 1994, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 7, 1995.

No significant hazards consideration comments received: No.

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

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: December 16, 1994; supplemented February 10, 1995 (TS 94–07).

Brief description of amendments: The amendments revise the technical specifications to reduce the high reactor power level setpoints when one or more main steam safety valves are inoperable and incorporate related changes.

Date of issuance: April 4, 1995.

Effective date: April 4, 1995.

Amendment Nos.: 196 and 187.

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

Date of initial notice in **Federal Register**: March 1, 1995 (60 FR 11140).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 4, 1995.

No significant hazards consideration comments received: None.

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

TU Electric Company, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: February 14, 1994 (Reference LAR 94–002, TXX–94008), as supplemented by letters dated May 17, 1994 (Reference TXX–94142), and April 3, 1995 (TXX–95098).

Brief description of amendments: The amendments revised Technical Specification (TS) 3/4.2.4, ''Quadrant Power Tilt Ratio,'' by replacing the existing TS and associated Bases concerning the quadrant power tilt ratio with a TS consistent with the improved Standard Technical Specifications (NUREG-1431).

Date of issuance: April 4, 1995.

Effective date: April 4, 1995.

Amendment Nos.: Unit 1—Amendment No. 36; Unit 2—Amendment No. 22.

Facility Operating License Nos. NPF–87 and NPF–89. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 20, 1994 (59 FR 37087).

The additional information contained in the supplemental letter dated April 3, 1995, was clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 4, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, TX 76019.

TU Electric Company, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: February 28, 1995 (Reference LAR 95–01).

Brief description of amendments: These amendments replace Surveillance Requirement (SR) 4.6.2.1b of Technical Specification (TS) 3/4.6.2, ''Depressurization and Cooling Systems—Containment Spray System,'' with the corresponding SR from NUREG–1431. Bases Section 3/4.6.2.1 ''Containment Spray System'' has also been revised to expand the detail consistent with the corresponding Bases from NUREG–1431. The SR, and its associated Bases, for confirming the performance of the containment spray pumps are changed by replacing the specific pump head and flow values with the general requirement that the pumps provide the required head at the flow test point while the specific required values are moved to the Comanche Peak Steam Electric Station Technical Requirements Manual.

Date of issuance: April 6, 1995.

Effective date: April 6, 1995.

Amendment Nos.: Unit 1—Amendment No. 37; Unit 2—Amendment No. 23.

Facility Operating License Nos. NPF–87 and NPF–89. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 6, 1995 (60 FR 12255).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 6, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, TX 76019.

TU Electric Company, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: April 22, 1994 (LAR 94–011, TXX–94116).

Brief description of amendments: These amendments changed Surveillance Requirement 4.7.1.2 of Technical Specification 3/4.7.1.2 ''Auxiliary Feedwater System,'' for the operational test frequency of the motor driven and turbine driven pumps from ''at least once per 31 days on a STAGGERED TEST BASIS'' to ''at least once per 92 days on a STAGGERED TEST BASIS.'' This change is consistent with ASME Section XI requirements and Generic Letter 93–05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operations.''

Date of issuance: April 7, 1995.

Effective date: April 7, 1995.

Amendment Nos.: Unit 1—Amendment No. 38; Unit 2—Amendment No. 24.

Facility Operating License Nos. NPF–87 and NPF–89. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 3, 1994 (59 FR 39598).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 7, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, TX 76019.

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

Date of application for amendment: December 9, 1994, as supplemented on January 27, 1995.

Brief description of amendment: The amendment revises the Technical Specification Surveillance Requirement 4.6.1.2.a and its associated Bases. The change defers the requirement to perform the Type A Containment Integrated Leak Rate Test until Refuel 8 (October 1996), in conjunction with the exemption to 10 CFR Part 50, Appendix J.

Date of issuance: April 5, 1995.

Effective date: April 5, 1995.

Amendment No.: 98.

Facility Operating License No. NPF–30. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** March 1, 1995 (60 FR 11141).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 5, 1995.

No significant hazards consideration comments received: No.

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

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: December 12, 1994.

Brief description of amendment: The amendment clarifies the surveillance requirements for verifying the correct required position for the valves in the auxiliary feedwater system.

Date of issuance: April 3, 1995.

Effective date: April 3, 1995, to be implemented within 30 days of issuance.

Amendment No.: 85.

Facility Operating License No. NPF–42. The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 18, 1995 (60 FR 3677).

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

No significant hazards consideration comments received: No.

Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621.

Wisconsin Public Service Corporation, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: April 11, 1994, as supplemented on November 30, and December 22, 1994, and March 3, 1995.

Brief description of amendment: The amendment revises Kewaunee Nuclear Power Plant (KNPP) Technical Specification (TS) 3.1.f, ''Minimum Conditions for Criticality,'' and its associated basis, by specifying that the moderator temperature coefficient (MTC) shall be no greater than 5.0 pcm/°F when at or below 60% rated thermal power and shall be zero or negative when above 60% rated thermal power. Additionally, the MTC shall be no less negative than -8 pcm/°F for 95% of the cycle time at full power. The amendment also incorporates required actions to be implemented, if the MTC specification is not met.

Date of issuance: April 3, 1995.

Effective date: April 3, 1995.

Amendment No.: 117.

Facility Operating License No. DPR–43. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** September 28, 1994 (59 FR 49442).

The November 30, and December 22, 1994, and March 3, 1995, submittals, provided clarifying information and expanded the basis portion of the TS, but did not change the initial no significant hazards consideration determination.

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

No significant hazards consideration comments received: None.

Local Public Document Room location: University of Wisconsin Library Learning Center, 2420 Nicolet Drive, Green Bay, Wisconsin 54301.

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

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

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

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

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

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

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

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By May 26, 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).

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

Date of application for amendments: March 29, 1995.

Brief description of amendments: The amendments change the Technical Specifications by revising the periodicity of the channel functional test of the turbine driven auxiliary feedwater pump from quarterly to each refueling outage.

Date of issuance: April 14, 1995.

Effective date: April 14, 1995.

Amendment Nos.: 161 and 149.

Facility Operating License Nos. DPR–39 and DPR–48. The amendments revised the Technical Specifications.

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

The Commission's related evaluation of the amendments, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated April 14, 1995.

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

Local Public Document Room location: Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085.

NRC Project Director: Robert A. Capra.

Dated at Rockville, Maryland, this 19th day of April 1995.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 95–10127 Filed 4–25–95; 8:45 am]

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Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission has issued a revision to a guide in its Regulatory Guide Series. This series has been developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the Commission's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits and licenses.

Revision 3 of Regulatory Guide 1.118, ''Periodic Testing of Electric Power and Protection Systems,'' describes a method acceptable to the NRC staff for complying with the Commission's regulations with respect to the periodic testing of the electric power and protection systems.

Comments and suggestions in connection with items for inclusion in guides currently being developed or improvements in all published guides are encouraged at any time. Written comments may be submitted 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.

Regulatory guides are available for inspection at the Commission's Public Document Room, 2120 L Street, NW., Washington, DC. Copies of issued guides may be purchased from the Government Printing Office at the current GPO price. Information on current GPO prices may be obtained by contracting the Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013–7082, telephone (202) 512–2249. Issued guides may also be purchased from the National Technical Information Service on a standing order basis. Details on this service may be obtained by writing NTIS, 5285 Port Royal Road, Springfield, VA 22161.

(5 U.S.C. 552(a))

Dated at Rockville, Maryland, this 12th day of April 1995.