Room: 315
Program: This meeting will review applications for projects in Interpretive Research: Humanities Studies of Medicine, submitted to Division of Research Programs, for projects beginning after July 1, 1995.

2. Date: January 27, 1995
Time: 8:30 a.m. to 5:00 p.m.
Room: 315
Program: This meeting will review applications for projects in Interpretive Research: Humanities Studies of Technology, Industry and Architecture, submitted to the Division of Research Programs, for projects beginning after July 1, 1995.

3. Date: January 30, 1995
Time: 8:30 a.m. to 5:00 p.m.
Room: 315
Program: This meeting will review applications for projects in Interpretive Research: History and Philosophy of Science, submitted to the Division of Research Programs, for projects beginning after July 1, 1995.

David C. Fisher,

Advisory Management Committee Officer.

[FR Doc. 95–11 Filed 1–3–95; 8:45 am]

BILLING CODE 7536–01–M

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–261]

Carolina Power & Light Company; H.R. Robinson Steam Electric Plant, Unit No. 2; Environmental Assessment and Finding of No Significant Impact

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. DPR–23 issued to Carolina Power & Light Company (the licensee) for operation of H.R. Robinson Steam Electric Plant, Unit No. 2 (HBR), located in Darlington County, South Carolina.

Environmental Assessment

Identification of Proposed Action

The proposed amendment would include provisions in Technical Specifications (TS) 5.3 and 5.4 which allow for the storage of fuel with an enrichment not to exceed 4.95 + 0.05 w/o U–235 in the new and spent fuel storage racks. The proposed action is in accordance with the licensee's application for amendment dated July 28, 1994.

The Need for Proposal Action

The proposed changes are needed so that the licensee can use higher fuel enrichment to provide the flexibility of extending the fuel irradiation and to permit operation for longer fuel cycles.

Environmental Impacts of the Proposed Action

The Commission has completed its evaluation of the proposed revisions to the TS. The proposed revisions would permit use of fuel enriched to a nominal 5.0 weight percent Uranium 235. The safety considerations associated with reactor operation with higher enrichment and extended irradiation have been evaluated by the NRC staff. The staff has concluded that such changes would not adversely affect plant safety. The proposed changes have no adverse effect on the probability of any accident. The higher enrichment, with fuel burnup to 60,000 megawatt days per metric ton Uranium, may slightly change the mix of fission products that might be released in the event of a serious accident, but such small changes would not significantly affect the consequences of serious accidents. No changes are being made in the types or amount of any radiological effluents that may be released offsite. There is no significant increase in the allowable individual or cumulative occupational radiation exposure.

With regard to potential nonradiological impacts of reactor operation with higher enrichment and extended irradiation, the proposed changes to the TS involve systems located with the restricted area, as defined in 10 CFR Part 20. They do not affect nonradiological plant effluents and have no other environmental impact.

The environmental impact of transportation resulting from the use of higher enrichment fuel and extended irradiation were published and discussed in the staff assessment entitled, ''NRC Assessment of the Environmental effect of Transportation Resulting from Extended Fuel Enrichment and Irradiation,'' dated July 7, 1988, and published in the **Federal Register** (53 FR 30355) on August 11, 1988. As indicated therein the environmental cost contribution of the proposed increase in the fuel enrichment and irradiation limits are either unchanged or may, in fact, be reduced from those summaries in Table S–4 as set forth in 10 CFR 51.52(c). Accordingly, the Commission concludes that there are no significant radiological environmental impacts associated with the proposed amendment.

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 non-radiological plant effluents and has no other environmental impact. Accordingly, the Commission concludes that there are no significant non-radiological environmental impacts associated with the proposed action.

Alternative to the Proposed Action

Since the Commission concluded that there are no significant environmental effects that would result from the proposed action, any other alternative would have equal or greater environmental impacts and need not be evaluated.

The principal alternative would be to deny the requested amendment. This would not reduce the environmental impact of plant operations and would result in reduced operational flexibility.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the Final Environmental Statement related to operation of HBR.

Agencies and Persons Consulted

The NRC staff reviewed the licensee's request and did not consult other agencies or persons.

Finding of No Significant Impact

The Commission has determined not to prepare an environmental impact statement for the proposed license amendments.

Based upon the foregoing environmental assessment, we conclude that the proposed action will not have a significant effect on the quality of the human environment.

For further details with respect to this action, see the application for amendments dated July 28, 1994, that is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555, and at the local public document room for the H.B. Robinson Steam Electric Plant, Unit No. 2, at Hartsville Memorial Library, 147 West College, Hartsville, South Carolina 29550.

Dated at Rockville, Maryland, this 28th day of December 1994.

For the Nuclear Regulatory Commission.

Byron L. Siegel,

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

[FR Doc. 95–125 Filed 1–3–95; 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 December 12, 1994, through December 21, 1994. The last biweekly notice was published on December 21, 1994 (59 FR 65809).

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 February 3, 1994, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 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.

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

Date of amendment requests: November 30, 1994

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

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

Standard 1 -- Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change relocates two tables of instrument response time limits from the TS to the UFSAR. The changes are in accordance with the guidance provided by the NRC in Generic Letter 93-08. The changes are administrative in nature and do not involve any modifications to plant equipment or affect plant operation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Standard 2 - Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change relocates two tables of instrument response time limits from the TS to the UFSAR. The changes are in accordance with the guidance provided by the NRC in Generic Letter 93-08. The changes are administrative in nature, do not involve any modifications to plant equipment and cause no change in the method by which any safety-related system performs its function. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3 - Does the proposed change involve a significant reduction in a margin of safety?

The proposed change relocates two tables of instrument response time limits from the TS to the UFSAR. The changes are in accordance with the guidance provided by the NRC in Generic Letter 93-08. The changes are administrative in nature, do not change or alter regulatory requirements and do not affect the safety analysis. Plant procedures contain response time testing acceptance criteria that reflect the reactor trip and ESFAS [engineered safety feature actuation system] response time limits in the tables being relocated from the TS into the UFSAR. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensees: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072-3999

NRC Project Director: Theodore R. Quay

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

Date of amendment requests: December 7, 1994

Description of amendment requests: The proposed amendment would change Table 4.3-1 of Technical Specification 3/4.3.1 to allow verification of the shape annealing matrix elements used in the Core Protection Calculators. This would provide the option to use generic shape annealing matrix elements in the Core Protection Calculators. Presently, cycle-specific shape annealing elements are determined during startup testing after each core reload. Use of a generic shape annealing matrix would eliminate approximately 2 to 3 hours of critical path work during startup after a refueling outage.

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

Standard 1 -- Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed Technical Specification amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The Technical Specification amendment provides the option to use generic shape annealing matrix elements in the Core Protection Calculators. The design basis of the Core Protection Calculators is to provide the DNBR [departure from nucleate boiling ratio] and linear heat rate trip functions for the Reactor Protection System so that the Specified Acceptable Fuel Design Limits on DNBR and fuel centerline melt are not exceeded during normal operation or Anticipated Operational Occurrences, and assist the Engineered Safety Features Actuation System in limiting the consequences of postulated accidents. The generic shape annealing matrix elements will be validated during startup testing and will meet the same acceptance criteria as the cycle specific shape annealing matrix elements. If the generic shape annealing matrix elements are not valid, cycle specific shape annealing matrix elements would be used in the Core Protection Calculators. This change will not affect the Core Protection Calculators capability to protect the plant by tripping the reactor, based on a conservative calculation of minimum DNBR and peak linear heat rate, to ensure that the Specified Acceptable Fuel Design Limits are not violated in the event of an Anticipated Operational Occurrence. Therefore, the generic shape annealing matrix elements will not affect the safety analysis, since there is no change to the design basis of the Core Protection Calculator System.

Standard 2 -- Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed Technical Specification amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. Since the generic shape annealing matrix elements will still have to meet the same acceptance criteria as the cycle specific shape annealing matrix elements, the Core Protection Calculators will still generate axial power shapes that fall within the required uncertainties. The Core Protection Calculators will still trip the reactor, based on a conservative calculation of minimum DNBR and peak linear heat rate, to ensure that the Specified Acceptable Fuel Design Limits are not violated in the event of an Anticipated Operational Occurrence.

Standard 3 -- Does the proposed change involve a significant reduction in a margin of safety?

The proposed Technical Specification amendment will not involve a significant reduction in a margin of safety. There is no reduction in the margin of safety, since the generic shape annealing matrix elements will still have to meet the same acceptance criteria as the cycle specific shape annealing matrix elements. Therefore, this change will not affect the design basis of the Core Protection Calculators. The Core Protection Calculators will still provide a reactor trip based on a conservative calculation of minimum DNBR and peak linear heat rate.

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

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

Attorney for licensees: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072-3999

NRC Project Director: Theodore R. Quay

Arizona Public Service Company, et al., Docket No. STN 50-529, Palo Verde Nuclear Generating Station, Unit No. 2, Maricopa County, Arizona

Date of amendment request: November 30, 1994

Description of amendment request: The proposed amendment would change the pressurizer code safety valve lift setting from 2500 psia to 2475 psia. The lift setting is being changed to permit Unit 2 to operate with up to 1500 plugged tubes in each steam generator.

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

Standard 1 -- Does the proposed change involve a significant increase in the probability or consequence of an accident previously evaluated?

The proposed Technical Specification amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. Chapters 6 and 15 of the [Palo Verde Nuclear Generating Station] PVNGS [Updated Final Safety Analysis Report] UFSAR have been reviewed to address the impact of these changes (1500 plugged tubes and a pressurizer code safety valve nominal lift setpoint of 2475 psia) on accident consequences. For most of the events that were previously analyzed in the UFSAR, the proposed change does not have a significant affect or adversely impact the accident analysis. For RCS [reactor coolant system] pressure peaking events, Loss of Condenser Vacuum (LOCV) and Feedwater Line Breaks (FLB), a new analysis was performed to justify the acceptability of the changes.

For the LOCV event (anticipated operational occurrence), the reanalysis determined that the peak RCS pressure, assuming 1500 plugged tubes and a pressurizer code safety valve nominal lift setpoint of 2475 psia, is 2728 psia. The maximum reactor coolant system (RCS) pressure reached for this event as described in UFSAR Section 15.2.3 is 2742 psia. Therefore, this change is bounded by the reference cycle (UFSAR analysis) and remains below the 110% (2750 psia) design pressure limit.

Several FLB scenarios are analyzed in support of PVNGS Unit 2 operation. The scenario with the highest system pressures is the large FLB with a loss of alternating current (LOAC). For the large FLB with a LOAC event (limiting fault event), assuming 1500 plugged tubes and a pressurizer code safety valve nominal lift setpoint of 2475 psia, is 2813 psia. The maximum RCS pressure reached for this event as described in UFSAR Section 15.2.8 is 2843 psia. The analysis shows that the RCS peak pressure for the large FLB with a LOAC (very low probability) event remains below the required value of 120% (3000 psia) of design pressure. Therefore, the analyses and reviews of the RCS pressure peaking events determined that the UFSAR design pressure limit is still bounding with this change. That is, the RCS design pressure limit will not be exceeded. Also, safety valves are accident mitigating devices and do not contribute to the probability of an event.

Standard 2 -- Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed Technical Specification amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The analyses and reviews show that the current licensing basis remains valid for this change (UFSAR design pressure limit is still bounding with this change). Safety valves are accident mitigating devices and do not contribute to the possibility of an accident. The pressurizer code safety valves are not manually or remotely operated, but are designed to automatically open to provide overpressure protection for pressure peaking events. The change in the pressurizer code safety valve setpoint to 2475 psia does not significantly increase the probability of a pressurizer code safety valve opening, since the pressure is still well above the Technical Specification Table 2.2-1 reactor trip setpoint of 2383 psia for high pressurizer pressure.

Standard 3 -- Does the proposed change involve a significant reduction in a margin of safety?

The proposed Technical Specification amendment does not involve a significant reduction in a margin of safety. The analyses and reviews show that the limits in the licensing and design basis are still valid with this change. The analyses show that the RCS peak pressure remains below the 110% (2750 psia) design pressure limit for the LOCV event and remains below the required value of 120% (3000 psia) of design pressure RCS peak pressure for the large FLB with a LOAC (very low probability) event. The analyses and reviews of the RCS pressure peaking events determined that the UFSAR design pressure limit is still bounding with this change. Therefore, the proposed Technical Specification amendment maintains the margin of safety to the design pressure limit.

The NRC staff has reviewed the licensees' analysis and, based on that 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: Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004

Attorney for licensees: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072-3999

NRC Project Director: Theodore R. Quay

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

Date of amendments request: November 16, 1994Description of amendments request: The proposed revision to the Technical Specifications (TS) would change the Technical Specification 3/4.6.2 to remove the specific instrumentation requirements for monitoring of the suppression chamber average water temperature. Also, the proposed revision would change the TS Bases 3/4.6.2 to indicate the methods that are acceptable for determining suppression chamber average water temperature.Proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

The proposed change maintains the same number of monitored locations from which an average suppression chamber water temperature can be derived, while making available additional valid RTD [resistance temperature detector] inputs from what was the redundant channel. No safety-related equipment, safety function or plant operation will be altered as a result of the proposed change. The SPTMS [suppression chamber temperature monitoring system] is neither an accident initiator nor does it provide any automatic accident mitigation function. The change does not affect the design, materials, or construction standards applicable to the suppression chamber average water temperature monitoring instrumentation.

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

The fundamental function and objective of the system is not affected by the proposed change. As stated above, no safety-related equipment, safety function or plant operations will be altered as a result of the proposed change. The change does not affect the design, materials, or construction standards applicable to the suppression chamber average water temperature instrumentation.

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

The proposed change allows the substitution of a qualified RTD already installed at a monitored location to insure the suppression chamber average water temperature remains valid. It does not involve any changes to the plant design or operation, therefore, no margins of safety, as defined by the plant's accident analyses, are impacted. Deletion of the defined instrument channels will not affect the ability to verify the suppression chamber ''average'' water temperature is being maintained below the maximum average temperatures required by the specification. This will insure the suppression chamber is Operable and able to perform its intended safety function.

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

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

Attorney for licensee: R. E. Jones, General Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: William H. Bateman

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

Date of amendment request: December 12, 1994

Description of amendment request: The requested change would revise the containment spray (CS) nozzle surveillance interval from 5 to 10 years.

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

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

The requested change extends the surveillance interval for performance of qualitative flow testing of the CS nozzles. A revision to this surveillance interval can in no way increase the probability of any accident previously evaluated.

Containment spray nozzle testing is not intended to track degradation of equipment by monitoring or trending performance. Rather, this surveillance constitutes a test of the passive design of the spray nozzles, i.e., it merely demonstrates whether the nozzles are or are not blocked or clogged. Based upon industry and plant-specific operating experience, a single failure rendering a significant number of nozzles inoperable as a result of blockage is considered highly unlikely. Since the reliability or functioning of the spray nozzles will not be affected by the revised surveillance interval, the consequences of any accident previously evaluated will not be increased. The requested change does not affect the physical design or operation of the plant, does not alter assumptions contained within the Updated Final Safety Analysis Report, and will not affect other Technical Specifications that preserve safety analysis assumptions. Therefore, operation of the facility in accordance with the requested change will not involve a significant increase in the consequences of any accident previously evaluated.

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

The requested change extends the surveillance interval for performance of qualitative flow testing of the CS nozzles. This change in the spray nozzle surveillance interval will not change or affect the physical plant or the modes of plant operation defined within the facility Operating License. This change does not involve the addition or modification of plant equipment, nor does it alter the design or operation of plant systems. Therefore, operation of the facility in accordance with the requested change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The requested change extends the surveillance for performance of qualitative flow testing of the CS nozzles. This revised surveillance interval will not change or otherwise influence the degree of operability assumed for the CS system within the plant safety analyses. As demonstrated by plant-specific and industry experience, an operational failure of the containment spray nozzles is considered highly unlikely. Since prior testing has demonstrated proper functioning of the CS spray nozzles, and operational single-failures are considered highly unlikely, a reduction in testing frequency should not affect the ability of the CS system to mitigate the affects of a large loss-of-coolant or steam release accident.

Therefore, operation of the facility in accordance with the requested change will not result in a significant reduction in the margin or 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: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550

Attorney for licensee: R. E. Jones, General Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: William H. Bateman

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

Date of amendment request: October 24, 1994

Description of amendment request: The proposed amendments would restructure the primary containment integrity and primary containment leakage technical specifications (TS) to reduce the repetition of those requirements contained in NRC regulations such as Appendix J to 10 CFR 50. The amendments also support proposed exemptions from Appendix J requirements related to the scheduling of containment integrated leak rate tests (CILRT). In addition to the restructuring and scheduling changes, the proposed amendments incorporate (1) the relocation of the list of primary containment isolation valves in accordance with Generic Letter 91-08, ''Removal of Component Lists from Technical Specifications,'' and (2) a revision of the interval for functional testing of hydrogen recombiners from 6 months to 18 months in accordance with Generic Letter 93-05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation.''

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

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

a. The relocation of Technical Specification 3/4.6.1.2, Primary Containment Leakage, and Surveillance Requirements 4.6.1.1.a, 4.6.4.3, and 4.6.6.1.d to specification 3/4.6.1.1, Primary Containment Integrity, as Surveillance Requirement 4.6.1.1.b continues to assure that Primary Containment leakage is maintained within the analyzed limit assumed for accident analysis by testing in accordance with 10 CFR part 50, Appendix J as modified by approved exemptions.

The requirement to be less than 0.75 L_a for as-left Type A test and less than 0.60 L_a for Type B and C tests prior to first unit startup following testing performed in accordance with 10 CFR part 50, Appendix J, as modified by approved exemptions, provides margin for degradation between tests and thus primary containment integrity is maintained during the time period between required leakage testing. The current Limiting Condition for Operation 3.6.1.2 in conjunction with Surveillance Requirements 4.6.1.2 basically require the same leakage limits as proposed Surveillance Requirement 4.6.1.1.b. The Limiting Condition for Operation (LCO) is required to be less than 1.0 L_a and is applicable during a fuel cycle for the Type A test. The LCO for Type B and C combined leakage total is currently required to be less than 0.60 L_a. The proposed Surveillance Requirement maintains the following:

1.The current LCO for Overall Containment leakage (as determined by a Type A test) and for the Type B and C combined leakage during the cycle by requiring overall containment leakage to be less than 1.0 L_a and Type B and C leakage total less than 0.60 L_a.

2. The associated limits specified in the current Action Statements are maintained by verifying Overall Containment leakage to be less than 0.75 L_a and Type B and C leakage total less than 0.60 L_a prior to startup from an outage in which the applicable leakage testing is conducted.

Therefore, there is no change to the consequences of an accident previously evaluated, because maintaining leakage within the analyzed limit assumed for accident analysis does not change either the onsite or offsite dose consequences resulting from an accident. In addition to this, containment leakage is not an accident initiator, so there is no effect on the probability of accident initiators. Thus there is no increase in the probability of an accident previously analyzed.

b. Relocation of Technical Specification table of Primary Containment Isolation Valves, Table 3.6.3-1, to the LaSalle UFSAR is an administrative change to remove the component list of Primary Containment Isolation Valves, Table 3.6.3-1, from the Technical Specifications. The Limiting Condition for Operation (LCO), 3.6.3, is being revised to define which components the LCO applies to. The wording of the revised LCO encompasses all of the components listed in the current Technical Specification Table 3.6.3. Removal of this component list does not change the probability of any accident initiators or change any other relevant initial assumptions. Also, there is no change to the consequences of an accident previously evaluated, because removing this list from Technical Specifications does not change either the onsite or offsite dose consequences resulting from the event. The component list will be controlled by an Administrative Procedure and can only be changed by the 10 CFR 50.59 change process with review and approval per the Onsite Review and Investigative Function. Therefore, there is no increase in either the probability or consequences of an accident previously evaluated.

c. The change in the functional test interval for the Drywell and Suppression Chamber Hydrogen Recombiner systems from ''once per 6 months'' to ''once per 18 months'' was determined by the NRC in NUREG 1366 and Generic Letter 93-05 to be acceptable by evaluation of the industry Licensing Event Reports (LERs) to assess the reliability of hydrogen recombiners. The conclusion was that the interval should be changed, because of the redundancy and apparent high reliability. A review of LaSalle LERs has shown only one LER that involved the operability of the hydrogen recombiner system and that was due to a Part 21 issue regarding circuit breaker environmental qualification. The breakers were replaced with qualified breakers. Therefore, the LaSalle Hydrogen Recombiner reliability is consistent with or better than that found by the NRC in determining this surveillance interval extension based on all LERs. Also, redundancy is the same as that assumed by the NRC; because, LaSalle has two hydrogen recombiner subsystems that are shared by Unit 1 and Unit 2. Both hydrogen recombiners subsystems are required to be Operable for either or both units in Operational Conditions 1 and 2. Based on LaSalle operating experience, the hydrogen recombiner subsystems are expected to continue to be demonstrated operable when the functional test is performed at an 18 month frequency.

Therefore, there is minimal or no change to the consequences of an accident previously evaluated, because at least one of the hydrogen recombiner subsystems is expected to be available to meet its design function to reduce the potential for hydrogen explosion or hydrogen burn in the primary containment. By preserving the integrity of the primary containment, there is no change to either the onsite or offsite dose consequences resulting from an accident. In addition to this, control of hydrogen concentration by use of a hydrogen recombiner subsystem is not an accident initiator, so there is no effect on the probability of accident initiators. Thus there is no significant increase in the probability of an accident previously analyzed.

d. The first exemption request is from the requirements of paragraph III.A.6(b) of Appendix J to allow LaSalle County Station Unit Two to return to or resume a Type A test schedule of three times in ten years (40 plus or minus 10 months). Due to consecutive failures, 10 CFR 50 Appendix J requires that Type A tests be performed every refueling outage on Unit Two until two consecutive Type A tests are satisfactory. 10 CFR Part 50 has an exemption process and is specified in 10 CFR Part 50.12(a), which states:

''The Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations of this part,...''

The exemption process requires showing that the granting of 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. Also, special circumstances are required to be present for the granting of an exemption. One of the special circumstances that would apply in this instance is 10 CFR part 50.12(a)(2)(ii) which states:

''Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule''.

This requires that it be shown that unacceptable containment leakage will be identified and corrected, by alternative methods. The alternative method is specifically Type B and C tests, which will identify any local penetration leakage. This is acceptable, because Type C test failures have been the cause for failures of as-found Type A tests in the LaSalle Unit 2 first, third, and fourth refueling outages.

Exceeding the allowable leakage rate during the performance of the Type A test is indicative of either a passive or a structural component that is leaking or that there is an inadequacy in the Local Leak Rate Test (Type B and C tests) program. When the failure of a Type A test is due to a passive or structural component, the only test for adequate repair would be the Type A test. For a Local Leak Rate Test program inadequacy, the Type A test would serve as a means of verification of the results of the test program. The Type A tests have not found new significant Type B or C tested local penetration leakage that has not been identified by Type B or C testing alone. Therefore, the LaSalle Local Leak Rate Test program is adequate to find and correct Type B and C containment penetration leakage.

When it is determined that Type A tests failed as a direct result of as-found Type B and C minimum path leakage penalty additions and not due to a non Type B or C tested components or structures, then performance of the Type A test more frequently as required by 10 CFR Part 50, Appendix J, due only to Type B and C test failures is redundant to the performance of Type B and C tests. Therefore, Type B or C tested penetration leakage that can be determined by Type B or C tests is evaluated and corrected, as applicable, to maintain overall containment leakage within limits, without an additional Type A test.

Primary Containment leakage which includes the minimum path Primary Containment Isolation Valve leakage is an assumption in any analyzed accident which could involve an offsite radioactive release. Because performance of Type B and C tests will find and allow correction/repair of leaking valves/penetrations, verification of as-found and as-left local leakage assures that Primary Containment leakage will be within the analyzed limit assumed for accident analysis.

Therefore, for this one-time exemption for LaSalle Unit 2, there is little or no increase in the consequences of an accident previously evaluated involving the dose previously calculated either onsite or offsite at the site boundary due to any analyzed accident. In addition to this, containment leakage is not an accident initiator, so there is no effect on the probability of accident initiators. Thus there is no significant increase in the probability of an accident previously analyzed.

e. The request for a partial exemption from paragraph III.D of Appendix J to 10 CFR 50 involves a deletion of the requirement to perform the third Type A test for each 10-year service period during the shutdown for the 10-year plant inservice inspections. There is no significant benefit in coupling these two surveillances (i.e., the Type A test and the 10-year ISI program). Each of the two surveillances is independent of the other and provides assurance of different plant characteristics. The Type A test assures the required leak-tightness for the reactor containment building be less than Appendix J acceptance criteria. This demonstrates compliance with the guidelines of 10 CFR Part 100 based on the assumptions used in the UFSAR which conform to NRC Safety Guide 4. The 10-year ISI program provides assurance of the integrity of the plant structures, systems, and components in compliance with 10 CFR 50.55(a). There is no safety-related concern necessitating their coupling to the same refueling outage. As a result, this change cannot increase the consequences (i.e., offsite dose) of any accident previously evaluated. Furthermore, since the decoupling of the test schedules has no affect on the test's effectiveness, decoupling their schedules will not increase the probability of an accident.

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

a. Technical Specification 3/4.6.1.2, Primary Containment Leakage, and Surveillance Requirements 4.6.1.1.a, 4.6.4.3, and 4.6.6.1.d are being relocated to specification 3.4.6.1.1, Primary Containment Integrity, as Surveillance Requirement 4.6.1.1.b. The proposed Surveillance Requirement 4.6.1.1.b assures that Primary Containment leakage is maintained within the analyzed limit assumed for accident analysis by testing in accordance with 10 CFR part 50, Appendix J as modified by approved exemptions. Primary containment leakage is an assumption in accident analyses, and is maintained by both the current specifications and the proposed specification. The leakage does not cause an accident and no new failure modes are created. Therefore this request for exemption does not create the possibility of a new or different kind of accident from any accident previously evaluated.

b. This is an administrative change to control the list of Primary Containment Isolation Valves outside the LaSalle Unit 1 and Unit 2 Technical Specifications. The administrative controls provided to control this component list assure that the design and operation of the plant will continue to be in accordance with the UFSAR, Facility License and the associated Technical Specifications. Therefore, the possibility of a new or different kind of accident from any previously evaluated is not created.

c. The change in the functional test interval for the Drywell and Suppression Chamber Hydrogen Recombiner systems from ''once per 6 months'' to ''once per 18 months'' is based on good equipment performance on a 6 month frequency. The expected outcome of the 18 month surveillances, based on the low failure rate at a six month frequency, is to show the hydrogen recombiner subsystems Operable. This system is for mitigating the consequences of an accident that causes generation of hydrogen and oxygen in the primary containment. No new failure modes are created by this change in surveillance frequency. Therefore, the possibility of a new or different kind of accident from any previously evaluated is not created.

d. The first exemption is from the requirements of paragraph III.A.6(b) of Appendix J to allow LaSalle County Station Unit Two to return to or resume a Type A test schedule of three times in ten years (40 plus or minus 10 months). Containment leakage testing, including both Type B and C testing and Type A testing as specified in the LaSalle County Station Safety Analysis Report were evaluated in Section 6.2.6 of Safety Evaluation Report, NUREG-0519, and found to be acceptable. Since Type B and C testing will find and verify correction of penetration leakage when Type B and C test as-found penalties are specifically what caused the failure of the as-found Type A tests, then Type B and C testing will provide adequate assurance of the continued integrity of the Primary Containment without increasing the frequency of Type A tests. As a result, the Primary Containment will continue [to] be maintained as designed and previously evaluated.

Based on this, the requirement of two acceptable as-found Type A tests prior to returning to the Appendix J paragraph III.D frequency of three times in ten years (40 plus or minus 10 months) is not necessary to assure that the primary containment remains within the analyzed leakage limits. Containment leakage is an assumption for the dose consequences of accident analyses, and not an accident initiator. Also, no new failure modes are created by this exemption. Therefore this Amendment does not create the possibility of a new or different kind of accident.

e. The request for a partial exemption from paragraph III.D of Appendix J to 10 CFR 50 involves a deletion of the requirement to perform the third Type A test for each 10-year service period during the shutdown for the 10-year plant inservice inspections. The proposed exemption does not involve any change to the plant design or operation. As discussed above, this change cannot increase the consequences of any accident previously evaluated. As a result, no new failure modes are created. Therefore, this proposed change cannot create the possibility of any new or different kind of accident from any accident previously evaluated.

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

a. Technical Specification 3/4.6.1.2, Primary Containment Leakage, and Surveillance Requirements 4.6.1.1.a, 4.6.4.3, and 4.6.6.1.d are being relocated to specification 3/4.6.1.1, Primary Containment Integrity, as proposed Surveillance Requirement 4.6.1.1.b. The proposed Surveillance Requirement 4.6.1.1.b continues to assure that Primary Containment leakage is maintained within the analyzed limit assumed for accident analysis by testing in accordance with 10 CFR part 50, Appendix J as modified by approved exemptions.

As stated in 1)a. above, the proposed Surveillance Requirement 4.6.1.1.b maintains the acceptance criteria and limits for continued operation of the current specification for primary containment leakage. Therefore, the margin of safety is not reduced by this change. Also, the proposed addition of a definition for the maximum allowable primary containment leakage rate assures that the margin of safety is maintained.

The leakage limits for MSIVs and hydrostatically tested valves are maintained by relocating the current surveillance requirements to specification 3/4.6.3, with the acceptance criteria of the current specification retained. Thus preserving the current margin of safety by maintaining the leakage rates as assumed in the accident analyses.

b. The Limiting Condition for Operation for Technical Specification 3.6.3, Primary Containment Isolation Valves, is revised by this Technical Specification change to specifically define the components to which the LCO applies. Therefore, removal of Technical Specification Table 3.6.3-1, which lists the specific components to which the LCO applies does not change the scope or applicability of the specification. The component list will be controlled administratively with any changes to the list made in accordance with the 10 CFR 50.59 change process. Therefore, this is an administrative change only and there is no reduction in the margin of safety.

c. The change in the functional test interval for the Drywell and Suppression Chamber Hydrogen Recombiner systems from ''once per 6 months'' to ''once per 18 months'' is based on good equipment performance on a 6 month frequency. The expected outcome of the 18 month surveillances, based on the low failure rate at a six month frequency, is to show the hydrogen recombiner subsystems Operable. The change in frequency has no affect on the hydrogen or oxygen generation assumptions or the recombination rate of the hydrogen recombiner subsystems. Therefore, the margin of safety is not reduced or changed by this surveillance interval change.

d. The first exemption is from the requirements of paragraph III.A.6(b) of Appendix J to allow LaSalle County Station Unit Two to return to or resume a Type A test schedule of three times in ten years (40 plus or minus 10 months). The limit of total leakage determined from Type B and C tests will remain the same, providing a margin of 40 percent to the maximum allowable containment leakage rate (L_a) at the design basis accident pressure specified in proposed Technical Specification definition of L_a. This 40 percent is as specified by 10 CFR Part 50, Appendix J. In addition to this, administrative guidelines have been set for each penetration/valve, so that any abnormal leakage will be corrected by adjustment or repair as needed. Any postponement of repairs is based on a technical evaluation and then only if the total Type B and Type C leakage is maintained at less than 0.60 L_a. Repairs will be required to restore the leakage rate to less than the administrative limit at the next refueling outage.

This request for exemption is based the fact that Type B and C testing minimum path leakage rate penalties are the direct cause of the failure of as-found Type A tests. The leakage through Type B and C tested penetrations is best measured and corrected via a local leak test. Therefore, verification of an adequate margin of safety is assured by conducting Type B and C tests, and not another increased frequency Type A test.

e. The request for a partial exemption from paragraph III.D of Appendix J to 10 CFR 50 involves a deletion of the requirement to perform the third Type A test for each 10-year service period during the shutdown for the 10-year plant inservice inspections. The proposed exemption does not change the acceptance criteria that must be met for inservice inspections, does not relax the condition of containment that must be met prior to plant restart, and does not change the requirements that must be met between plant refueling outages. Therefore, the proposed change does not result in a significant reduction in the margin of safety.

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

Local Public Document Room location: Public Library of Illinois Valley Community College, Rural Route No. 1, Oglesby, Illinois 61348

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

NRC Project Director: Robert A. Capra

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: November 21, 1994

Description of amendment request: The proposed amendment would revise the Technical Specifications to allow a one-time extension of the allowed outage time for an inoperable reserve source of offsite power.

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 an accident previously evaluated.

The proposed changes will extend the allowed outage time for the Reserve source of off-site power, on a one time basis, to allow the installation of high speed protective relays on the unit system auxiliary transformers which will increase the level of protection from ground faults on the low voltage (secondary) side of the transformers. Operation of Zion, Units 1 and 2, in accordance with the proposed requirements will not affect the initiators or precursors of any accident previously evaluated. Operation in accordance with the proposed requirements will not increase the likelihood that a transient initiating event will occur because transients are initiated by equipment malfunction and/or catastrophic system failure. As a result, the probability of occurrence of accidents previously evaluated is not significantly increased.

During the [system auxiliary transformer] SAT outage, power to the shut down unit will be provided by backfeeding off-site power through the unit main power transformers and the UAT to supply the unit non-essential 4-KV service buses. Emergency on-site power will be available to the shut down unit from at least one unit specific [emergency diesel generator] EDG when fuel is in the reactor core. This will ensure that at least one train of Residual Heat Removal (RHR) will have an emergency source of AC power at all times. RHR Train A is powered by ESF bus 149(249) which can be energized by the 1B(2B) EDG during a loss of off-site power. RHR Train B is powered by bus 148(248) which can be energized by the 1A(2A) EDG. Because the 'O' EDG must be operable for the operating unit, it will also be available to energize the Division 7 ESF bus on the shut down unit. The 'O' EDG can supply buses 147 and 247 simultaneously if the need should arise during an emergency.

Power to the operating unit (opposite unit) will be provided by the SAT and the UAT in the normal at-power configuration. Emergency on-site will be provided by the two unit specific EDGs (A and B) and the common 'O' EDG. In accordance with the proposed requirements, the Reserve source of off-site power will not be removed from service unless all three EDGs are operable and the normal source of off-site power is operable. Administrative controls will be in place to limit activities in the switchyard that could impact the reliability of the remaining source of off-site power to the unit.

The Zion PRA was used to compare the impact of extending the action time versus the impact of manual reactor shutdown on core damage probability. The PRA result concluded that the risk of continuing to operate the operating unit for an additional 11 days with the shutdown unit's SAT out of service is not significantly greater than the risk of manually shutting down the operating unit at the expiration of the current 72 hour action statement and is not significant when compared to the total core damage probability in a year.

The revised surveillance requirements will provide additional assurance that redundant sources of power are maintained operable while the reserve source of off-site power is unavailable. The ability to safely shut down the operating unit and mitigate the consequences of all accidents previously evaluated will be maintained. The reserve source of off-site power is not relied upon in any design basis accident. Therefore, based on the previous discussion, the proposed changes do not involve a significant increase in 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 the Technical Specifications 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 analysis. No safety related equipment or function will be altered as a result of the proposed changes. Also, the procedures governing normal plant operation and recovery from an accident are not changed by the proposed Technical Specification changes. The proposed changes will extend the allowed outage time for the Reserve source of off-site power, on a one-time basis, to allow the installation of high speed protective relays on the unit system auxiliary transformers which will increase the level of protection from ground faults on the low voltage (secondary) side of the transformers. The addition of the high speed relaying has been evaluated pursuant to 10 CFR 50.59, and no unreviewed safety questions were identified.

Requirements will be modified to require additional assurance that the remaining off-site source of AC power and the on-site source of emergency (emergency diesel generators) are OPERABLE. 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.

The proposed changes will extend the allowed outage time for the reserve source of off-site power, on a one-time basis, to allow for installation of high speed protective relays on the unit system auxiliary transformers which will increase the level of protection from ground faults on the low voltage (secondary) side of the transformers.

During the SAT outage, power to the operating unit (opposite unit) will be provided by the unit SAT and the UAT in the normal configuration. Emergency on-site power will be provided by the two unit specific EDGs (A and B) and the common 'O' diesel generator. Because the accident analyses take no credit for offsite power availability, this temporary degradation will not impact the analysis results.

No safety system setpoints are changed by this proposal. There is no impact on any physical design margins, and no analytical results are affected by this change. The revised surveillance requirements will provide additional assurance that redundant sources of power are maintained operable while the Reserve source of off-site power is unavailable.

Based on the above discussion, 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 amendment request involves no significant hazards consideration.

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

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NRC Project Director: Robert A. Capra

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

Date of amendment request: October 5, 1994

Description of amendment request: The proposed amendment would (1) revise primary coolant system (PCS) pressure-temperature (P-T) limits, power-operated relief valve (PORV) setting limits, and primary coolant pump starting limits to accommodate reactor vessel fluence for an additional 4 effective full power years (up to 2.192 x 10^{19}nvt). The existing P-T limit curves are calculated for a fluence of 1.8 x 10^{19} could be reached as early as March 1, 1995; (2) require the high pressure safety injection (HPSI) pumps to be ''rendered incapable of injection into the PCS'' when the PCS is below 300°F, rather than the existing requirement to render both HPSI pumps ''inoperable'' when the PCS is below 260°F. This change supports the assumption in the P-T limit analyses that HPSI injection would not occur below 300°F; and (3) establish a more restrictive limit on pressurizer heatup rate to achieve consistency between design assumptions and technical specification (TS) limits. The limit in the existing TS is less restrictive than used in design calculations. Neither the design heatup rate nor the TS heatup rate limit is achievable with installed equipment.

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

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

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

The revision of the Primary Coolant Pump [PCP] starting limits, PCS P-T curves, and PORV setting limits would not cause any changes to the capability or operation of plant systems that would affect the probability of occurrence or consequences of an accident. These revisions simply update the existing requirements to account for additional reactor vessel fluence.

The reduction of the allowable pressurizer heatup rate would have no effect on operation of the plant. The current limit is physically unobtainable with installed equipment. The proposed change better aligns the Technical Specification limits with the design analysis. The change in the pressurizer heatup rate limit will not increase the probability or consequences of an accident.

Requiring the HPSI pumps to be operable when above 325°F, rather than when above 300°F does not affect the probability or consequences of any accident previously evaluated. Neither the existing 300°F requirement nor the proposed 325°F requirement has an analytical base. This requirement was recently changed from 325°F to 300°F simply for uniformity. With the revised P-T limit analysis requirement to assure that inadvertent HPSI injection will not occur below 300°F, it is necessary to revert to the former limit of 325°F to provide time to transition between these two contrasting HPSI pump requirements.

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

The revised specifications, PCP starting limits, PCS P-T limits, pressurizer heatup rate, PORV setting limits, and HPSI pump restrictions, all are directly related to, and intended to prevent, a previously analyzed event, failure of the Reactor Coolant Pressure Boundary. Revision of these limits would not create the possibility of a new or different kind of accident.

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

The revised PCP starting limits, PCS P-T limits, and PORV setting limits are calculated using a similar methodology as the limits which they replace. Therefore they provide the same margin of safety.

The revised pressurizer heatup rate reduces the currently allowable limit which is in the direction of increased margin of safety. Since there is no equipment installed which would cause either the existing or the proposed limit to be reached, there will be no change on the operation of the plant equipment. Therefore reducing the limit on the pressurizer heatup rate will not involve a significant reduction in the margin of safety.

Requiring the HPSI pumps to be operable when above 325°F, rather than when above 300°F does not involve a significant reduction in any margin of safety. Neither the existing 300°F requirement nor the proposed

325°F requirement has an analytical base. This requirement was recently changed from 325°F to 300°F simply for uniformity. With the revised P-T limit analysis requirement to assure that inadvertent HPSI injection will not occur below 300°F, it is necessary to revert to the former limit of 325°F to provide time to transition between these two contrasting HPSI pump requirements.

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

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

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

NRC Project Director: John N. Hannon

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

Date of amendment request: November 8, 1994

Description of amendment request: The proposed amendment revises technical specifications (TSs) associated with requirements for performing the containment integrated leak rate test (ILRT). The proposed change describes the ILRT test frequency by referencing the test frequency requirements included in 10 CFR Part 50, Appendix J. The existing specifications paraphrase the Appendix J requirements, but include differences that result in interpretation problems.

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 revises Technical Specification 4.4.1.1.4 to reference the testing frequency requirements of 10 CFR 50, Appendix J, and to state that NRC approved exemptions to the applicable regulatory requirements are permitted. The current requirements of TS 4.4.1.1.4 paraphrase the requirements of Section III.D.1.(a) of Appendix J. The proposed administrative revision simply deletes the paraphrased language and directly references Appendix J. No new requirements are added, nor are any existing requirements deleted. An approved exemption to Section III.D.1.(a) of Appendix J would not necessarily affect the requirements of TS 4.4.1.1.4, unless the proposed clarification phrase permitting the use of approved exemptions is added. Any specific changes to the requirements of Section III.D.1(a) will require a submittal from Entergy Operations under 10CFR50.12 and subsequent review and approval by the NRC prior to implementation. The proposed change is stated generically to avoid the need for further TS changes if different exemptions are approved in the future.

The proposed change, in itself, does not affect reactor operations or accident analysis and has no radiological consequences. The change provides clarification so that TS changes will not be necessary in the future to correspond to applicable NRC approved exemptions from the requirements of Appendix J. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed change provides clarification to a specification which paraphrases a codified requirement. Since the proposed amendment would not change the design, configuration or method of operation of the plant, it would not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change is administrative and clarifies the relationship between the requirements of TS 4.4.1.1.4, Appendix J, and any approved exemptions to Appendix J. It does not, in itself, change a safety limit, an LCO, or a surveillance requirement on equipment required to operate the plant. The NRC will directly approve change proposed exemption to III.D.1.(a) of Appendix J prior to implementation. Therefore, this change does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: William D. Beckner

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-321 and 50-366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

Date of amendment request: December 2, 1994

Description of amendment request: The proposed amendments would replace Appendix B, ''Environmental Technical Specifications'' with an Environmental Protection Plan (Nonradiological) and revise the Operating Licenses to reflect these changes. The proposed changes are administrative in nature, altering only the format and location of programmatic controls and procedural details relative to nonradiological environmental monitoring.

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

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

The proposed changes to the Environmental Technical Specifications (ETS) are administrative in nature, altering only the format and location of programmatic controls and procedural details relative to nonradiological environmental values. The proposed Environmental Protection Plan (EPP) (Nonradiological) contains the programmatic controls now residing in the ETS, with appropriate plant procedures serving as implementing documents. The proposed changes to the operating licenses are also administrative in nature and change the Appendix B reference from ETS to EPP. Compliance with applicable regulatory requirements will be maintained. In addition, the proposed changes do not alter the conditions or assumptions in any of the accident analyses. Therefore, these proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes to the ETS do not involve any change to the configuration or method of operation of any plant equipment. These proposed changes are administrative in nature and consist of replacing the ETS with an EPP. The proposed changes to the operating licenses are also administrative in nature and change the Appendix B reference from ETS to EPP. Accordingly, no new failure modes have been identified for any plant system or component important to safety nor has any new limiting single failure been identified as a result of the proposed changes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes to the ETS relate primarily to matters involving recordkeeping, reporting, and administrative procedures or requirements. No significant change in the type or quantity of any effluent release will result from this action. These changes replace the ETS with an EPP. The proposed EPP contains the programmatic controls now residing in the ETS, with appropriate plant procedures serving as implementing documents to ensure compliance with applicable regulatory requirements. The proposed changes to the operating licenses are also administrative in nature and change the Appendix B reference from ETS to EPP. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

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

NRC Project Director: Herbert N. Berkow

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: November 21, 1994

Description of amendment request: The proposed amendment would eliminate the Main Steam Isolation Valve (MSIV) - Leakage Control System (LCS) including the primary containment isolation valves associated with the MSIV - LCS, along with increasing the allowable MSIV leakage rates.

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 proposed changes to TS Section 3.6.1.2 do not involve a change to structures, components, or systems that would affect the probability of an accident previously evaluated. The TS limits for MSIVs are increased from 46 scf per hour for all four main steam lines to less than or equal to 100 scf per hour for any one MSIV and a combined maximum pathway leakage rate of less than or equal to 300 scf per hour for all four main steam lines. The consequences of an accident are affected as discussed in this section.

The proposed changes to TS Section 3.6.1.4 eliminate the Main Steam Isolation Valves (MSIVs) Leakage Control System (LCS) requirements from the TS. As described in Section 6.7 of the FSAR, the LCS is manually initiated in about 20 minutes following a design basis Loss of Coolant Accident (LOCA). Since the LCS is operated only after an accident has occurred, these proposed changes have no effect on the probability of an accident.

Since MSIV leakage and operation of the LCS are included in the radiological analysis for the design basis LOCA as described in Section 15.6.5 of the FSAR, the proposed changes do not affect the precursors of other analyzed accidents. Analysis of the effects of the proposed changes do, however, result in acceptable radiological consequences for the design basis LOCA previously evaluated in Section 15.6.5 of the FSAR.

SSES, Units 1 and 2 have an inherent MSIV leakage treatment capability as discussed below. We propose to use the drain lines associated with the main steam lines and main turbine condenser as an alternative to the guidance in Regulatory Guide 1.96, ''Design of Main Steam Isolation Valve Leakage Control System For Boiling Water Nuclear Power Plants'', Revision 0, May 1975, for MSIV leakage treatment. If approved, we will incorporate this alternate method in the appropriate operational procedures and Emergency Operating Procedures.

The Boiling Water Reactor Owners' Group (BWROG) has evaluated the availability of main steam system piping and main condenser alternate pathways for processing MSIV leakage, and has determined that the probability of a near coincident LOCA and a seismic event is much smaller than for other plant safety risks. Accordingly, this alternate MSIV leakage treatment pathway is available during and after a LOCA. Nevertheless, the BWROG has also determined that main steam piping and main condenser design are extremely rugged, and the design requirements applied to SSES Unit 1 and Unit 2 main steam system piping and main condenser contain substantial margin, based on the original design requirements. Therefore, the alternate treatment method has been evaluated for its capability to mitigate the consequences of a LOCA, and has been evaluated to assure its availability considering a seismic event.

In order to determine the capability of the main steam piping and main condenser alternate treatment pathway, the BWROG has reviewed earthquake experience data on the performance of non-seismically designed piping and condensers during past earthquakes. The data is summarized in General Electric (GE) Report, ''BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems,'' NEDC 31858P, Revision 2, submitted to the NRC by BWROG letter dated October 4, 1993. This study concluded that the possibility of a failure that could cause a loss of steam or condensate in Boiling Water Reactor (BWR) main steam piping or condensers in the event of a design basis (i.e., safe shutdown) earthquake is highly unlikely, and that such a failure would also be contrary to a large body of historical earthquake experience data, and thus unprecedented.

A verification has been performed of the seismic adequacy of the Unit 1 and Unit 2 main steam piping and main condenser consistent with the guidelines discussed in Section 6.7 of NEDC-31858P, Revision 2, to provide reasonable assurance of the structural integrity of these components. An evaluation, including the walkdown report outliers, ''MSIV Leakage Alternate Treatment Method Seismic Evaluation,'' for Unit 1 and Unit 2, is attached. The results of the evaluation clearly demonstrate that the MSIV Leakage Alternate Treatment Method meets the intent of 10CFR100 Appendix A, with regards to seismic qualification. Except for the requirement to establish a proper flow path from the MSIVs to the condenser, the proposed method is passive and does not require any additional logic control and interlocks. The method proposed for MSIV leakage treatment is consistent with the philosophy of protection by multiple barriers used in containment design for limiting fission product release to the environment.

A plant-specific radiological analysis has been performed in accordance with NEDC-31858P, Revision 2, to assess the effects of the proposed increase to the allowable MSIV leakage rate in terms of control room and off-site doses following a postulated design basis LOCA. This analysis utilizes the hold-up volumes of the main steam piping and condenser as an alternate method for treating the MSIV leakage. As discussed earlier, there is reasonable assurance that the main steam piping and condenser remain intact following a design basis earthquake. The radiological analysis uses standard conservative assumptions for the radiological source term consistent with Regulatory Guide (RG) 1.3, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-Of-Coolant Accident for Boiling Water Reactor, Revision 2, dated April 1974.

The analysis results demonstrate that dose contributions from the proposed MSIV leakage rate limit of 100 scfh per steam line, not to exceed a total of 300 scfh for all four main steam lines, and from the proposed deletion of the LCS, result in an insignificant increase to the LOCA doses previously evaluated against the regulatory limits for the off-site doses and control room doses contained in 10CFR100 and 10CFR50, Appendix A, General Design Criterion (GDC) 19, respectively. The off-site and control room doses resulting from a LOCA are discussed in Section 15.6.5 of the FSAR. The off-site and control room doses resulting from a LOCA associated with the proposed changes are the sum of LOCA doses evaluated in the power uprate revision to the design basis DBA-LOCA calculation (EC-RADN-1009) and the additional doses calculated using the alternate MSIV leakage treatment method. Enclosure 3 [of application dated November 21, 1994] summarizes the off-site and control room doses and compares the alternate treatment method doses to the original MSIV-LCS treatment method doses.

The 30-day whole body doses at the Low Population Zone (LPZ) did not change and remained at .37 rem for the alternate treatment method. The 30-day control room whole body doses increased slightly from .38

rem to .76 rem for the alternate treatment method. The increase in control room dose is not significant since the revised doses are well below the regulatory limits, i.e., .76 rem calculated versus the limit of 5 rem in the control room. The two-hour whole body dose at the Exclusion Area Boundary (EAB) decreased slightly from 2.47 rem to 2.217 rem.

The 30-day thyroid dose at the LPZ increased from 30.4 rem for the MSIV-LCS treatment method to 41.74 rem for the alternate treatment method. This increase is not significant since the revised dose of 41.74 rem is well within the regulatory limit of 300 rem. The two-hour thyroid dose at the EAB decreased slightly from 127.8 rem to 125.61 rem. The 30-day control room thyroid dose increased from 14.19 rem for the MSIV-LCS treatment method to 18.55 rem for the alternate treatment method. The increased control room thyroid dose is not significant since the revised dose remains well below the regulatory limit of 30 rem.

The 30-day control room beta dose increased insignificantly from 12 rem for the MSIV-LCS treatment method to 12.17 rem for the alternate treatment method, remaining a small fraction relative to the limit of 75 rem.

In summary, the proposed changes discussed above do not result in a significant increase in the radiological consequences of a LOCA when the same assumptions and methods specified in the FSAR are used, recognizing that radiological consequences calculated in the FSAR and for these proposed changes are significantly higher than those using more realistic assumptions and methods. Nevertheless, the calculated off-site and control room doses resulting from a LOCA remain well below the regulatory limits.

The proposed change to TS Table 3.6.3-1 deletes the LCS valves from the list of primary containment isolation valves. This proposed change is consistent with the proposed deletion of the LCS. The LCS lines that are connected to the main steam piping are welded and/or capped closed to assure primary containment integrity is maintained. The welding and post weld examination procedures will be in accordance with American Society of Mechanical Engineers (ASME) Code, Section III requirements. These welds and/or caps will be periodically tested as part of the Containment Integrated Leak Rate Test (CILRT). This proposed change does not involve an increase in the probability of equipment malfunction previously evaluated in the FSAR. In fact, this proposed change reduces the probability of equipment malfunction since, upon implementation of these proposed changes, the plant will be operated with less primary containment isolation valves subjected to postulated failure. This proposed change has no effect on the consequences of an accident since the LCS lines will be welded and/or cap closed, thus assuring that the containment integrity, isolation and leak test capability are not compromised.

The proposed change to TS Table 3.8.4.2.1-1 deletes the LCS motor operated valves from the list of ''Motor Operated Valves Thermal Overload Protection - Continuous.'' The proposed change has no effect on the probability or consequences of an accident since the valves are eliminated and not performing a safety function.

Therefore, as discussed above, the proposed changes do not involve a significant increase in the probability or consequences from any accident previously evaluated.

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

As stated in Section I, the proposed changes do not involve a change to structures, components, or systems that would affect the probability of an accident previously evaluated , nor would these changes create any new or different kind of accident from any previously evaluated. The proposed changes will introduce and take credit for a new level of operational performance for existing plant systems and components to mitigate the consequences of the accident. The effect on this equipment has been evaluated and found to provide an acceptable level of reliability resulting in the required level of protection. This conclusion is based on the evaluation performed in NEDC 31858P, Revision 2, and the plant specific seismic evaluation provided in the Enclosure 2 [of application dated November 21, 1994], ''MSIV Leakage Alternate Treatment Method Seismic Evaluation.'' The Leakage Control System has been installed to direct any leakage past the MSIVs during the LOCA; acting after the accident has occurred. The resulting consequences of the evaluated accidents have been affected as discussed in Section I resulting in no significant increase in the probability or consequences of said accident. Therefore, reliance on different equipment than previously assumed to mitigate the consequences of an accident does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The BWROG evaluated MSIV performance and concluded that MSIV leakage rates up to 200 scfh per valve will not inhibit the capability and isolation performance of the MSIVs to effectively isolate the primary containment. Implementation of the proposed changes does not result in modifications which could adversely impact the operability of the MSIVs. The LOCA has been analyzed using the main steam piping and main condenser as a treatment method to process MSIV leakage at the proposed maximum rate of 100 scfh per main steam line, not to exceed 300 scfh total for all four main steam lines. Therefore, the proposed TS Section 3.6.1.2 change to increase the allowed MSIV leakage rate does not create any new or different kind of accident from any accident previously evaluated.

The proposed TS Section 3.6.1.4 change to eliminate the LCS does not create the possibility of a new or different kind of accident from any accident previously evaluated because the removal of the LCS does not affect any of the remaining SSES Unit 1 and Unit 2 systems, and the LOCA has been re-analyzed using the proposed alternate method to process MSIV leakage. The associated proposed change to delete the LCS isolation valves from TS Table 3.6.3-1 and Table 3.8.4.2.1-1 does not create the possibility of a new or different kind of accident. The affected main steam piping will be welded and/or capped closed to assure that the primary containment integrity, isolation, and leak testing capability are not compromised. The affected LCS motor operated valves will be eliminated so their thermal overloads will not need to be bypassed.

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

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

The proposed change to TS Section 3.6.1.2 to increase the MSIV allowable leakage does not involve a significant reduction in the margin of safety. As discussed in the current Bases for TS Section 3/4.6.1.2, the allowable leak rate limit specified for the MSIVs is used to quantify a maximum amount of leakage assumed to bypass primary containment in the LOCA radiological analysis. Accordingly, results of the re-analysis supporting these proposed changes are evaluated against the dose limits contained in 10CFR100 for the off-site doses, and 10CFR50, Appendix A, GDC 19, for the control room doses. As discussed above, sufficient margin relative to the regulatory limits is maintained even when assumptions and methods (e.g., RG 1.3) that are considered highly conservative relative to more realistic assumptions and methods are used in the analysis.

Results of the radiological analysis demonstrate that the proposed changes do not involve a significant reduction in the margin of safety. Whole body doses, in terms of margin of safety, are insignificantly reduced by .38 rem in the control room. The margin of safety remains constant for the LPZ whole body dose or actually increases by .253 rem for the EAB whole body dose. The margin of safety for thyroid dose category is reduced by 11.34 rem at the LPZ and 4.36 rem in the control room. The margin of safety is found to increase for the EAB thyroid dose by 2.19 rem. The margin of safety for beta dose is insignificantly reduced by .17 rem in the control room. The reductions in the margin of safety are not significant since the revised calculated doses are highly conservative yet remain well below the regulatory limits, and therefore, a substantial margin to the regulatory limits is maintained.

The proposed change to eliminate the LCS from TS Section 3.6.1.4 does not reduce the margin of safety, in fact, the overall margin of safety is increased. The function of the LCS for MSIV leakage treatment will be replaced by alternate main steam drain lines and condenser equipment. This treatment method is effective in reducing the dose consequences of MSIV leakage over an expanded operating range compared to the capability of the LCS and will, thereby, resolve the safety concern that the LCS will not function at MSIV leakage rates higher than the LCS design capacity. Except for the requirement to establish a proper flow path from the MSIVs to the condenser, the proposed method is passive and does not require any new logic control and interlocks. This proposed method is consistent with the philosophy of protection by multiple barriers used in containment design for limiting fission product release to the environment. Furthermore, as previously identified, based on the evaluations discussed in NEDC-31858P, Revision 2, and the seismic evaluation provided in the Enclosure 2 [of application dated November 21, 1994] report, ''MSIV Leakage Alternate Treatment Method Seismic Evaluation,'' the design of the MSIV leakage alternate drain pathway, meets the intent of the 10CFR100, Appendix A requirement for seismic qualification. Therefore, the proposed method is highly reliable and effective for MSIV leakage treatment.

The revised calculated LOCA doses remain within the regulatory limits for the off-site and the control room. Therefore, the proposed method maintains a margin of safety for mitigating the radiological consequences of MSIV leakage for the proposed TS leakage rate limit of 100 scfh per main steam line, not to exceed a total of 300 scfh for all four main steam lines.

The proposed change to delete LCS isolation valves from TS Table 3.6.3-1 and Table 3.8.4.2.1-1 does not reduce the margin of safety. Welded and/or capped closure of the LCS lines assures that the primary containment integrity and leak testing capability are not compromised. These welds and/or caps will be periodically leak tested as part of the CILRT. The LCS motor operated valves will be eliminated so their thermal overloads will not need to be bypassed. Therefore, the proposed deletion of the LCS isolation valves does not involve a 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

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

Date of amendment request: November 18, 1994

Description of amendment request: The proposed change would revise the Reactivity Control System Technical Specification Limiting Conditions for Operation for boration flow paths and charging pumps by reducing the number of operable charging pumps required for boron addition in Mode 4 from two to one.

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 or malfunction of equipment important to safety previously evaluated.

The Emergency Core Cooling System (ECCS) requirements assume that only one charging pump will be available below 350°F without single failure considerations on the bases of the stable reactivity condition of the reactor and limited core cooling requirements. Therefore, the Mode 4 Applicability has been deleted from LCOs 3.2.1.2 and 3.2.1.4, and was added to LCOs 3.2.1.1 and 3.2.1.3 consistent with the requirements of LCO 3.5.3.

The current Bases for the Unit 2 Technical Specification for boration system flow paths via the charging pumps supports the use of a similar LCO for Salem Unit 1.

The limitation for a maximum of one centrifugal charging pump to be operable when the RCS temperature is less than or equal to 312°F has been added to LCO 3.1.2.3 for clarity and is consistent with the Cold Overpressure Protection (POPS) analysis and the requirements of Technical Specification 3.5.3.

The requirements for Boric Acid Transfer Pump operability are adequately addressed in Technical Specifications 3.1.2.1 and 3.1.2.2 which specify the boron injection flow paths to be operable and the components required to perform this function. This includes the availability of the transfer pumps to meet this Technical Specification requirement.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated in the UFSAR.

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

As discussed in response to Question 1 above, the proposed amendment to the number of charging pumps required to be operable in Mode 4 is consistent with the current Technical Specification requirements for the ECCS LCO and the POPS. The current bases for the Unit 2 Technical Specification for boration system flow paths via the charging pumps supports the use of a similar LCO for Salem Unit 1. The requirements for Boric Acid Transfer Pump operability for Unit 1 are adequately addressed in Technical Specifications 3.1.2.1 and 3.1.2.2 which specify the boron injection flow paths to be operable and the components required to be available to perform this function including the transfer pumps. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Will not involve a significant reduction in a margin of safety. The proposed amendment to the number of charging pumps required to be operable in Mode 4 will not result in any changes to the assumptions or conditions for the current ECCS analysis and POPS analysis. The current bases for the Unit 2 Technical Specification for boration system flow paths via the charging pumps supports the use of a similar LCO for Salem Unit 1 (i.e., the Bases are essentially the same). The requirements for Boric Acid Transfer Pump operability for Unit 1 are adequately addressed in Technical Specifications 3.1.2.1 and 3.1.2.2 which specify the boron injection flow paths to be operable and the components required to be available to perform this function including the transfer pumps. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John F. Stolz

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

Date of amendments request: December 19, 1994Description of amendments request: The proposed change to Table 3.7-3 of the Technical Specifications includes the revision to the main steam safety valve (MSSV) setpoint tolerance from plus or minus 1 percent to plus or minus 3 percent and modifies the bases to 3/4.7.1.1 to increase the relieving capacity of the MSSVs to at least 12,984,660 pounds per hour which corresponds to approximately 112 percent of total secondary steam flow at 100 percent rated thermal power. In addition, modifications to Table 3.7-1 are proposed to reduce the allowable power range neutron flux high setpoints for multiple inoperable steam generator safety valves. The proposed amendment includes an editorial correction to Bases 3/4.7.1.2 to indicate required auxiliary feedwater flow at ''1133 psia'' rather than ''1133 psig.''

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 license amendment does not involve a significant increase in the

probability or consequences of an accident previously evaluated.

These proposed changes to the Farley Technical Specifications do not result in a condition where the design, material and construction standards of the MSSVs that were applicable prior to the proposed change are altered. The valves will continue to function as designed. All applicable safety analyses have been reviewed, evaluated or reanalyzed and all applicable safety criteria continue to be met. No accident sequences are altered because of the proposed amendment. The radiological consequences for the Steam Generator Tube Rupture were reanalyzed and 10 CFR 100 criteria continue to be met. All other FSAR radiological analyses remain bounding. Analyses have been performed to justify the proposed high nuclear flux setpoint changes. All acceptance criteria for these analyses continue to be met. Therefore, the proposed amendment does not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

The MSSVs continue to have the required pressure relieving capacity to ensure that system design pressure remains below 110% of shell design pressure. The proposed changes are not accident initiators nor do they create any new accident scenarios or any new limiting single failures. The ability of the MSSVs to respond to an accident condition is not impaired by the proposed changes. The proposed high nuclear flux setpoints for multiple valves out of service ensure all applicable safety criteria for accident analyses are met. No new accident scenarios are created by these proposed changes. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Acceptance criteria for accident analysis continue to be met. Radiological consequences for the affected Chapter 15 analysis remain within 10 CFR 100 acceptance criteria. No safety limits or safety system setpoint requires modification due to the proposed changes. The current secondary side over-pressure limit of 100% of steam generator shell design pressure is not violated. Analysis for the high nuclear flux setpoints have verified that there is no reduction in margin for the events analyzed. Therefore, there is not 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: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

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

NRC Project Director: William H. Bateman

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

Date of amendment request: September 9, 1994

Description of amendment request: The proposed amendment would revise Technical Specification 3/4.8.1 and its associated Bases to improve emergency diesel generator reliability and availability. Several surveillance requirements would be revised or eliminated, and guidance provided in Regulatory Guide 1.9, Revision 3, and Generic Letter 93-05 would be incorporated.

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

The proposed changes do not involve a significant hazards consideration because operation of Callaway Plant with these changes would not:

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

These proposed changes do not involve a change in the operational limits or physical design of the emergency power system. Emergency diesel generator operability and reliability will continue to be assured while minimizing the number of required emergency diesel generator starts. Also, emergency diesel generator reliability will be enhanced by minimizing service test conditions which can lead to premature failures.

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

These proposed changes do not involve a change in the operational limits or physical design of the emergency power system. The performance capability of the emergency diesel generator will not be affected. Emergency diesel generator reliability and availability will be improved by the implementation of the proposed changes. There is no actual impact on accident analysis.

3. Involve a Significant Reduction in the Margin of Safety.

These proposed changes do not involve a change in the operational limits or physical design of the emergency power system. The performance capability of the emergency diesel generator will not be affected. Emergency diesel generator reliability and availability will be improved by the implementation of the proposed changes. No margin of safety is reduced.

Based on the above discussions, it has been determined that the requested technical specification revision does not involve a significant increase in the probability or consequences of an accident or other adverse condition over previous evaluations; or create the possibility of a new or different kind of accident or condition over previous evaluations; or involve a significant reduction in a margin of safety. The requested license amendment does not involve a significant hazards consideration.

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

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

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

NRC Project Director: Leif J. Norrholm

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

Date of amendment request: September 9, 1994

Description of amendment request: The proposed amendment would revise Technical Specification 3.8.2.1 and 3.8.2.2, 125-volt D.C. busses for battery bank and chargers and provides for the installation of swing chargers during the next refueling outage. Technical Specifications 3.8.3.1 and 3.8.3.2 would be revised to address the 120-volt A.C. Vital Busses.

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 to the Technical Specifications do not involve a significant hazards consideration because operation of Callaway Plant in accordance with these changes would not:

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

These proposed Technical Specification changes do not involve any hardware changes nor do they affect the probability of any event initiators. There will be no change to normal plant operating parameters or accident mitigation capabilities. There will be no increase in the consequences of any accident or equipment malfunction.

2) Create the possibility for accident or malfunction of equipment of a different type than previously evaluated in the FSAR.

The proposed Technical Specification changes do not involve any design changes nor are there any changes to the method by which any safety-related plant system performs its safety function. The normal manner of plant operation is unaffected. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes.

Involve a significant reduction in the margin of safety.

There will be no affect [SIC] on the manner in which safety limits or limiting safety system settings are determined, nor will there be any effect in those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on DNBR limits, F_Q, F-delta-H, LOCA PCT, peak local power density or any other margin of safety.

Based on the information presented above, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and does [SIC] not involve a significant hazards consideration.

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

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

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

NRC Project Director: Leif J. Norrholm

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

Date of amendment request: August 27, 1993

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to be consistent with recent revisions to 10 CFR Part 20 and 10 CFR 50.36a. Administrative changes are 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. The proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated. The changes as proposed consist of revisions to the Technical Specifications to meet new regulatory requirements as contained in 10CFR20 and 10CFR50.36a, and other related changes of an administrative nature. There is no change in the types and amounts of effluents released, nor will there be any increase in individual or cumulative occupational radiation exposures. None of the changes proposed will affect any plant hardware, plant design, safety limit settings, or plant system operation, and therefore do not modify or add any initiating parameters that would significantly increase the probability or consequences of any previously analyzed accident.

2. The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes as proposed do not physically alter the plant nor do they change the operation of the plant.

3. The proposed change does not involve a significant reduction in the margin of safety. The changes will not increase the amount or types of effluents that may be released offsite, nor do they significantly increase individual or cumulative occupational radiation exposures. These changes will not alter any of the requirements or responsibilities for protection of the public and/or employees against radiation hazards.

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

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

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

NRC Project Director: Walter R. Butler

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

Date of amendment request: March 31, 1994

Description of amendment request: The proposed amendment would modify the requirements for avoidance and protection from thermal hydraulic instabilities to be consistent with the Boiling Water Reactor (BWR) Owner's Group long-term solution Option 1-D described in the Licensing Topical Report, ''BWR Owner's Group Long-Term Stability Solutions Licensing Methodology, NEDO-31960 June 1991'' and NEDO-31960, Supplement 1, dated March 1992. NEDO-31960 and NEDO-31960, Supplement 1, were accepted by the NRC staff in a letter to L.A. England (BWR Owner's Group) dated July 12, 1993.

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

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The implementation of BWR Owner's Group long term stability solution Option 1-D at Vermont Yankee does not modify the assumptions contained in the existing accident analysis. The use of an exclusion region and the operator actions required to avoid and minimize operation inside the region do not increase the possibility of an accident. Conditions of operation outside of the exclusion region are within the analytical envelope of the existing safety analysis. The operator action requirement to exit the exclusion region upon entry minimizes the possibility of an oscillation occurring. The actions to drive control rods and/or to increase recirculation flow to exit the region are maneuvers within the envelope of normal plant evolutions. The flow biased scram has been analyzed and will provide automatic fuel protection in the event of an instability. Thus, each proposed operating requirement provides defense in depth for protection from an instability event while maintaining the existing assumptions of the accident analysis.

2. The proposed amendment will not create the possibility of a new or different kind of accident from an accident previously evaluated. As stated in 1), the proposed operating requirements either mandate operation within the envelope of existing plant operating conditions of force specific operating maneuvers within those carried out in normal operation. Since operation of the plant with all of the proposed requirements are within the existing operating basis, an unanalyzed accident will not be created through implementation of the proposed change.

3. The proposed amendment will not involve a significant reduction in a margin of safety. Each of the proposed requirements for plant thermal hydraulic stability provides a means for fuel protection. The combination of avoiding possible unstable conditions and the automatic flow biased reactor scram provides an in depth means for fuel protection. Therefore, the individual or combination of means to avoid and suppress an instability supplements the margin of safety.

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

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

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

NRC Project Director: Walter R. Butler

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

Date of amendment request: November 29, 1994

Description of amendment request: Virginia Electric and Power Company plans to insert fuel assemblies containing fuel rods, guide thimble tubes, instrumentation tubes, and mid-span grids fabricated with Westinghouse Electric Corporation's (Westinghouse's) advanced zirconium alloy material, ZIRLO, into the Surry Units 1 and 2 reactors, beginning with Cycle 14 at each unit. In the current fuel design, these components are fabricated from Zircaloy-4.

Because the Technical Specifications define the fuel rod cladding material as Zircaloy-4, implementation of this material change requires changes to the Technical Specifications. Technical Specification 5.3.A.1 is being modified to allow the use of either Zircaloy-4 or ZIRLO fuel rod cladding, and an additional reference for the calculation of the heat flux hot channel factor for loss-of-coolant-accident evaluations of fuel with ZIRLO cladding is being defined in Technical Specification 6.2. The use of the ZIRLO fabricated guide thimble tubes, instrumentation tubes, and mid-span grids does not require changes to the Technical Specifications.

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

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

1. Involve a significant increase in the probability or consequence of an accident previously evaluated. The Surry fuel assemblies containing fuel rods, guide thimble tubes, instrumentation tubes and mid-span grids fabricated with ZIRLO alloy meet the same fuel assembly and fuel rod design bases as the current fuel assemblies fabricated with Zircaloy-4 components. In addition, the 10 CFR 50.46 criteria will be applied to the fuel rods, guide thimble tubes, instrumentation tubes and mid-span grids fabricated with ZIRLO alloy. The use of these fuel assemblies will not result in a change to the Surry Units 1 and 2 reload design and safety analysis limits. The ZIRLO alloy is similar in chemical composition to Zircaloy-4, and also has physical and mechanical properties similar to those of Zircaloy-4. Thus the cladding integrity is maintained and the structural integrity of the fuel assembly is not affected. The ZIRLO clad fuel rods improve corrosion resistance and dimensional stability. Since the dose predictions in the safety analyses are not sensitive to the fuel rod cladding material changes as specified in this report, the radiological consequences of accidents previously evaluated in the safety analyses remain valid. Therefore, neither the probability of occurrence nor the consequences of any accident previously evaluated is significantly increased.

2. Create the possibility of a new or different kind of accident from any accident previously identified, since the Surry Units 1 and 2 fuel assemblies containing fuel rods, guide thimble tubes, instrumentation tubes and mid-span grids fabricated with ZIRLO alloy will satisfy the same design bases used for previous fuel regions containing Zircaloy-4 components. Since the original design criteria are being met, the fuel rods, guide thimble tubes, instrumentation tubes and mid-span grids fabricated with ZIRLO alloy will not be initiators for any new accident. Applicable design and performance criteria will continue to be met and no single failure mechanisms have been created. In addition, the use of these fuel assemblies does not involve any alteration to plant equipment or procedures which would introduce any new or unique operational modes or accident precursors. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

3. Involve a significant reduction in a margin of safety. The Surry Units 1 and 2 fuel assemblies containing fuel rods, guide thimble tubes, instrumentation tubes and mid-span grids fabricated with ZIRLO alloy do not change the Surry Units 1 and 2 reload design and safety analysis limits. The use of fuel assemblies containing fuel rods, guide thimble tubes, instrumentation tubes and mid-span grids fabricated with ZIRLO alloy will take into consideration the normal core operating conditions allowed in the Technical Specifications. For each cycle reload core these fuel assemblies will be specifically evaluated using approved reload design methods and approved fuel rod design models and methods. This will include consideration of the core physics analysis peaking factors and core average linear heat rate effects. Analyses or evaluations will be performed each cycle to confirm that the 10 CFR 50.46 criteria will be met for the use of fuel with fuel rods, guide thimble tubes, instrumentation tubes and mid-span grids fabricated with ZIRLO alloy. Therefore, the margin of safety as defined in the Bases to the Surry Units 1 and 2 Technical Specifications is not significantly reduced.

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

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

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

NRC Project Director: Mohan C. Thadani, Acting

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

Date of amendment request: December 2, 1994

Description of amendment request: The proposed amendment would revise Kewaunee Nuclear Power Plant (KNPP) Technical Specification (TS) 3.2 by eliminating the requirements for the charging pumps, high concentration boric acid in the boric acid storage tanks (BASTs), the boric acid transfer pumps, and boric acid heat tracing. Changes to TS 3.3 and Table TS 3.5.3 are also being proposed to add requirements associated with the emergency core cooling system (ECCS) accumulators, remove the requirements associated with the boric acid storage tanks, and to increase the minimum required boron concentration in the refueling water storage tank (RWST). Additionally, the surveillance requirements involving the BASTs, associated valves and heat tracing located in Table TS 4.1-1, Table TS 4.1-2 and Section 4.5 would be eliminated. Supporting analysis for the limiting design basis accident conditions have been performed using the proposed minimum RWST boron concentration of 2400 ppm.

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

Significant Hazards Determination for Proposed Changes to Technical Specification (TS) 3.2 and Table TS 3.5-3.

The proposed changes were reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed changes will not:

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

Neither the charging pumps, the high concentration boric acid, the BASTs, the boric acid transfer pumps nor the boric acid heat tracing system are accident initiators. Therefore, a change to these systems will not significantly increase the probability of an accident previously evaluated. The effect of a reduction in initial safety injection boron concentration on the accident analysis was evaluated. The limiting accidents were the Large-Break Loss-of-Coolant Accident

(LOCA) and the Steam Line Break (SLB) event. A decrease in the initial safety injection boron concentration from 20,000 ppm to 2400 ppm will not adversely affect the Large or Small-Break Loss-of-Coolant Accident analysis because the evaluation models used in analyzing these accidents do not take credit for the high concentration boric acid stored in the BASTs. However, the evaluation models did take credit for boron in maintaining the long term post LOCA reactor core sub-critical. An analysis was performed which concluded that the inventory contained in the BASTs would not be required provided the minimum RWST boron concentration was increased to 2400 ppm. The SLB event is the other design basis event that could be affected by the proposed elimination of the high boron concentration BASTs as a source of safety injection fluid. Analyses have been performed which conclude that the BASTs are not required and that a minimum RWST boron concentration of only 1950 ppm is sufficient to provide adequate protection for the SLB event although 2400 ppm will be maintained to address post-LOCA subcriticality thus providing further safety margin. The results of these analyses indicate that the departure from nucleate boiling (DNB) design basis continues to be met. (A minimum Departure from Nucleate Boiling Ratio (DNBR) of 1.45 can be maintained throughout the event.) Finally, the containment pressure and temperature remains within the acceptable containment design limits. Since these criteria have been satisfied, there will be no adverse effect on the health and safety of the public and the consequences of any accident previously evaluated have not significantly increased.

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

Neither the charging pumps, the removal of the BASTs from initial SI pump injection, nor the elimination of both the boric acid transfer pumps and the boric acid heat tracing system as safety-related components would create the possibility of a new or different kind of accident from any accident previously evaluated.

Furthermore, the reactivity control function of the boron in the CVCS and SI systems is not being changed. Therefore, the proposed changes will not adversely affect the health and safety of the public or create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The reduction in the initial concentration of boron injected into the reactor coolant system for accident mitigation has been analyzed. These analyses conclude that all applicable criteria for a LOCA are satisfied. A decrease in the initial safety injection boron concentration from 20,000 ppm to 2400 ppm will not adversely effect the Large- or Small-Break Loss-of-Coolant Accident analysis because the evaluation models used in analyzing these accidents do not take credit for the high concentration boric acid stored in the BASTs. However, in order to maintain the long term post LOCA reactor core sub-critical, a minimum RWST boron concentration of 2400 ppm is required. To meet this requirement, the RWST boron concentration is being raised to 2400 ppm. All criteria of 10 CFR 50.46 can be achieved for both the Large or Small-Break LOCA with no BASTs and 2400 ppm boron in the RWST. Since all criteria of 10 CFR 50.46 are satisfied, there is no adverse effect on the health and safety of the public and there is not a significant reduction in the margin of safety for these casualties.

Since both the core response and the containment response can be limiting in the SLB event, both were considered in the boron concentration reduction analysis. This analysis concludes that a minimum RWST boron concentration of 1950 ppm is sufficient to provide adequate protection for the SLB event, although a 2400 ppm boron solution will be maintained to provide protection for the post LOCA concerns. Since the containment pressure and temperature remains within the acceptable containment design limits, and a minimum DNBR of 1.45 can be maintained throughout the event, there is not a significant reduction in the margin of safety for this event and therefore there is no adverse effect on the health and safety of the public.

These proposed changes involve the conversion of the TS to Word Perfect format now being used at WPSC. Minor typographical errors and format inconsistencies were corrected. These proposed changes are administrative in nature; accordingly, these proposed changes do not involve a significant hazards consideration.

Additionally, the proposed changes are similar to example C.2.e.(i) in 51 FR 7751. Example C.2.e.(i) states that changes which are purely administrative in nature; i.e., to achieve consistency throughout the Technical Specifications, correct an error, or a change in nomenclature, are not likely to involve a significant hazard.

Significant Hazards Determination for Proposed Changes to Table TS 4.1-1, ‘‘Minimum Frequencies for Checks, Calibrations and Test of Instrument Channels’’ and Table TS 4.1-2 ‘‘Minimum Frequencies for Sampling Tests’’

The proposed changes were reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed changes will 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 the margin of safety.

The above listed surveillance requirements insure BAST operability. The BASTs will no longer be relied upon as a source of boron for safety injection, and will serve no safety related function. Whether the BASTs are operable or not will have no effect on plant safety. Therefore, elimination of the surveillance requirements which insure BAST operability is possible without any adverse effect on the health and safety of the public and presents no significant hazards.

Significant Hazards Determination for Proposed Changes to Technical Specification TS 3.3 and Section 4.5.

The proposed changes were reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed changes will not:

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

Neither the RWST, the boron solution contained within the RWST nor valves SI-3, SI-4A/B are accident initiators. Therefore, a change to these systems will not significantly increase the probability of an accident previously evaluated. The effect of a reduction in initial Safety Injection boron concentration on the accident analysis was evaluated. The limiting accidents were the Large-Break Loss-of-Coolant Accident (LOCA) and the Steam Line Break (SLB) event. A decrease in the initial safety injection boron concentration from 20,000 ppm to 2400 ppm will not adversely effect the Large or Small-Break Loss-of-Coolant Accident analysis because the evaluation models used in analyzing these accidents do not take credit for the high concentration boric acid stored in the BASTs. However, the evaluation models did take credit for boron in maintaining the long term post LOCA reactor core sub-critical. An analysis was performed which concluded that the BASTs could be eliminated provided the minimum RWST boron concentration was increased to 2400 ppm. The SLB event is the other design basis event that could be affected by the proposed elimination of the high concentration BASTs as a safety-related source for reactivity control injection fluid. However, analyses have been performed which conclude that a minimum RWST boron concentration of only 1950 ppm is sufficient to provide adequate protection for the SLB event although 2400 ppm will be maintained to address post-LOCA subcriticality thus providing further safety margin. The results of these analyses indicate that the departure from nucleate boiling (DNB) design basis continues to be met. (A minimum Departure from Nucleate Boiling Ratio (DNBR) of 1.45 can be maintained throughout the event.) Furthermore, maintaining the suction of the SI pumps to the RWST with valves SI-4A or SI-4B open with power removed places the system in a normal SI sequence and eliminates the requirement to switch suction from the BASTs to the RWST. This eliminates a potential failure mechanism and increases the overall reliability of the ECCS system. Finally, the containment pressure and temperature remains within the acceptable containment design limits.

Since these criteria have been satisfied, there will be no adverse effect on the health and safety of the public and the consequences of any accident previously evaluated have not significantly increased.

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

This change to the Technical Specifications allows use of 2400 ppm boron for safety injection. SI pump suction would be directly from the RWST. This eliminates the necessity of shifting suction from the BASTs to the RWST, reducing the complexity of the operation. Since the pumps remain connected to the RWST throughout the injection phase, there is no possibility of a new or different kind of accident from any accident previously evaluated.

Neither the reduction in initial boron concentration for safety injection, nor the increase in the boron concentration in the RWST would create the possibility of a new or different kind of accident from any accident previously evaluated.

Lastly, the reactivity control function of the boron in the CVCS and SI systems is not being changed. Therefore, the proposed changes will not adversely affect the health and safety of the public or create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The change in concentration of boron injected into the primary system for accident mitigation has been analyzed. These analyses conclude that all applicable criteria for a LOCA are satisfied. A change in safety injection boron concentration to 2400 ppm will not adversely affect the Large or Small-Break LOCA analysis because the evaluation model codes used in analyzing these accidents did not take credit for boron. However, a minimum RWST boron concentration of 2400 ppm is required to maintain long term post LOCA reactor core sub-criticality. To meet this requirement, the RWST minimum boron concentration is being raised to 2400 ppm. All criteria of 10 CFR 50.46 can be achieved for both the Large or Small-Break LOCA with 2400 ppm boron in the RWST. Since all criteria of 10 CFR 50.46 are satisfied, there is no adverse effect on the health and safety of the public and there is not a significant reduction in the margin of safety for these casualties.

Since both the core response and the containment response can be limiting in the SLB event, both were considered in the boron concentration reduction analysis. Although a minimum RWST boron concentration of 1950 ppm is sufficient to provide adequate protection for the SLB event, a 2400 ppm boron solution will be maintained to provide protection for the post large break LOCA concerns. Since the containment pressure remains below the design pressure, and a minimum DNBR of 1.45 can be maintained throughout the event, there is not a significant reduction in the margin of safety for this event.

These proposed changes involve the conversion of the TS to Word Perfect format now being used at WPSC. Minor typographical errors and format inconsistencies were corrected. These proposed changes are administrative in nature; accordingly, these proposed changes do not involve a significant hazards consideration.

Additionally, the proposed changes are similar to example C.2.e.(i) in 51 FR 7751. Example C.2.e.(i) states that changes which are purely administrative in nature; i.e., to achieve consistency throughout the Technical Specifications, correct an error, or a change in nomenclature, are not likely to involve a significant hazard.

Significant Hazards Determination for Proposed Changes to Technical Specification (TS) Section 4.5 ''Emergency Core Cooling System and Containment Air Cooling System Tests.''

The proposed changes were reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed changes will 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 the margin of safety.

The above listed surveillance requirements insure BAST operability. The BASTs will no longer be relied upon as a source of boron for safety injection, and will serve no safety related function. Whether the BASTs are operable or not will have no effect on plant safety. Therefore, elimination of the surveillance requirements which insure BAST operability is possible without any adverse effect on the health and safety of the public and presents no significant hazards.

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

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

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P. O. Box 1497, Madison, Wisconsin 53701-1497.

NRC Project Director: Leif J. Norrholm

Previously Published Notices Of Consideration Of Issuance Of Amendments ToFacility 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. 50-368, Arkansas Nuclear One,Unit No. 2, Pope County, Arkansas

Date of amendment request: November 29, 1994

Brief description of amendment request: The proposed amendment would delete requirements to perform the full complement of steam generator surveillances as outlined in the technical specifications (TSs) when the steam generators are subjected to special inspections that are in addition to inspections that are required by the TSs.

Date of individual notice in the **Federal Register:** December 5, 1994 (59 FR 62416)

Expiration date of individual notice: January 4, 1995

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

Notice Of Issuance Of Amendments To Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental

Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

Arizona Public Service Company, et al., Docket No. STN 50-528, Palo Verde Nuclear Generating Station, Unit No. 1, Maricopa County, Arizona

Date of application for amendment: November 22, 1994

Brief description of amendment: The amendment adds a note to Technical Specification Table 3.7-2. The note allows continuous operation of Unit 1 during Cycle 5 at 100-percent maximum steady state power with one main steam safety valve inoperable per steam generator. This note applies only during the current fuel cycle (Cycle 5) for Unit 1.

Date of issuance: December 19, 1994

Effective date: December 19, 1994

Amendment No.: 87

Facility Operating License No. NPF-41: The amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: Yes (59 FR 61907, dated December 2, 1994). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by December 19, 1994, but stated that, if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of significant hazards consideration is contained in a Safety Evaluation dated December 19, 1994.

Attorney for licensee: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072-3999

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

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

Date of application for amendment: October 7, 1994

Brief description of amendment: The amendment revises the introduction to TS Section 6.9.3.3 to require the approved revision number for the referenced analytical methods to be listed in the Core Operating Limits Report. The methodology referenced in 6.9.3.3.b.f (XN-NF-82-49(A)) has been updated to clarify that all supplements are included. New methodologies ANF-89-151(A) and EMF-92-081(A) will be added to TS Section 6.9.3.3.b.

Date of issuance: December 12, 1994

Effective date: December 12, 1994

Amendment No.: 154

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

Date of initial notice in **Federal Register:** November 9, 1994 (59 FR 55868)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 12, 1994.No significant hazards consideration comments received: No

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

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

Date of application for amendment: February 18, 1994, as supplemented by letters dated June 3, 1994, November 1, 1994, December 2, 1994, December 14, 1994 and December 16, 1994.

Brief description of amendment: The amendment revises surveillance intervals for the Vapor Containment Sump Discharge Flow and Temperature Channel, the Loss of Power Undervoltage and Degraded Voltage Relays, and the Control Rod Protection System Trip to accommodate a 24-month refueling cycle. In addition it changes the trip setpoint for the Control Rod Protection System Trip. These revisions are being made in accordance with the guidance provided by Generic Letter 91-04, ''Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle.''

Date of issuance: December 20, 1994

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

Amendment No.: 179

Facility Operating License No. DPR-26: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 28, 1994 (59 FR 22003) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 20, 1994.No significant hazards consideration comments received: No

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

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

Date of application for amendments: July 19, 1994

Brief description of amendments: These amendments change Technical Specification 3.1.5 for each unit for the standby liquid control system (SLCS) to remove the operability requirement for the SLCS while the plant is in Operational Condition 5 (refueling) with any control rod withdrawn, and to delete the 18-month system surveillance requirement (Surveillance Requirement 4.1.5.d.3).

Date of issuance: December 20, 1994

Effective date: December 20, 1994

Amendment Nos.: 136 and 106

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

Date of initial notice in **Federal Register:** August 17, 1994 (59 FR 42344)The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 20, 1994.No significant hazards consideration comments received: No

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

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

Date of application for amendments: July 22, 1994

Brief description of amendments: The amendment removes the surveillance frequency details regarding 10 CFR Part 50, Appendix J, Types B and C testing from the Technical Specifications.

Date of issuance: December 19, 1994

]*Effective date:* December 19, 1994

Amendment Nos. 83 and 44

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

Date of initial notice in **Federal Register:** September 14, 1994 (59 FR 47180) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 19, 1994.No significant hazards consideration comments received: No

Local Public Document Room location: Pottstown Public Library, 500

High Street, Pottstown, Pennsylvania 19464.

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

Date of application for amendment: October 7, 1994

Brief description of amendment: The amendment revises Technical Specification (TS) 4.6E.4 and the associated Bases to establish that the manual cycling of reactor coolant system safety/relief valves (SRVs) during plant startups is to be accomplished within 12 hours after steam pressure and flow are adequate to perform the testing. TS 4.6E.4 had previously required that this testing be performed within 12 hours of continuous power operation at a reactor steam dome pressure of at least 940 psig. The amendment also makes several editorial changes to clarify the intent of TSs involving SRV testing and performance requirements.

Date of issuance: December 16, 1994

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

Amendment No.: 219

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

Date of initial notice in **Federal Register:** November 9, 1994 (59 FR 55889)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 16, 1994.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.

Public Service Electric & Gas Company, Docket Nos. 50-272 and 50-311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey Date of application for amendments: September 9, 1994

Brief description of amendments: The amendments revise the Technical Specification surveillance requirements regarding visual inspection of snubbers and are consistent with the guidance provided in Generic Letter 90-09, ''Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions.''

Date of issuance: December 12, 1994

Effective date: December 12, 1994

Amendment Nos. 161 and 142

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

Date of initial notice in **Federal Register:** November 9, 1994 (59 FR 55889)The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 12, 1994.No significant hazards consideration comments received: No

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

Public Service Electric & Gas Company, Docket Nos. 50-272 and 50-311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey Date of application for amendments: March 28, 1994, as supplemented June 1, 1994, and August 24, 1994

Brief description of amendments: The amendments revise the sustained degraded voltage relay trip setpoint and the allowable value due to changes in the switchyard configuration.

Date of issuance: December 14, 1994

Effective date: December 14, 1994

Amendment Nos. 162 and 143

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

Date of initial notice in **Federal Register:** June 8, 1994 (59 FR 29633) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 14, 1994.No significant hazards consideration comments received: No

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

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

Date of application for amendments: October 7, 1993 (TS 313)

Brief description of amendments: The changes include the addition of the high range primary containment radiation monitors and recorders and the wide range gaseous effluent radiation recorder and monitor, which were installed at the Browns Ferry facility in response to NUREG 0737 ''Clarification of TMI Action Plan Requirements'' and GL 83-36, into the technical Specifications (TS) for Units 1 and 3. Similar changes to the Unit 2 TS were issued previously (Amendment Nos. 125 and 171). The amendment also clarifies that the high range primary containment radiation recorders and monitors are both part of the instrument loop. The amendment contains administrative typographical changes which provide consistency for the TS tables and footnotes for Units 1 and 3.

Date of issuance: December 21, 1994

Effective Date: December 21, 1994

Amendment Nos.: 214, 230, 187

Facility Operating License Nos. DPR-33, DPR-52 and DPR-68: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 22, 1993 (58 FR 67863)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 21, 1994.No significant hazards consideration comments received: None

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

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: November 11, 1994, as supplemented by letter dated November 16, 1994.

Brief description of amendments: The proposed amendment would modify Comanche Peak Steam Electric Station Technical Specification Table 4.8-1, ''Diesel Generator Test Schedule,'' by excluding two valid failures of the Unit 2 Train B diesel generator from contributing towards an accelerated test schedule.

Date of issuance: December 9, 1994

Effective date: December 9, 1994

Amendment Nos.: Unit 1 - Amendment No. 33; Unit 2 - Amendment No. 19

Facility Operating License Nos. NPF-87 and NPF-89. The amendments revised the Technical Specifications.Public comments requested as to proposed no significant hazards consideration: Yes (59 FR 69399, dated November 23, 1994). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by December 23, 1994, but stated that, if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendments.

The Commission's related evaluation of the amendments, finding of exigent circumstances, and final determination of no significant hazards consideration is contained in a Safety Evaluation dated December 9, 1994.

Local Public Document Room location: University of Texas at Arlington library, Government

Publications/Maps, 702 College, P.O. Box 19497, Arlington, Texas 76019.

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

Date of application for amendments: March 29, 1994

Brief description of amendments: These amendments revise Point Beach Nuclear Plant Technical Specification 15.3.2, ''Chemical and Volume Control System,'' by eliminating the necessity for high concentration boric acid and removing the operability requirements for the associated heat tracing. The basis for Section 15.3.2 and applicable surveillances in Table 15.4.1-2 are also revised to support the above changes.

Date of issuance: December 12, 1994

Effective date: Date of issuance, to be implemented within 45 days.

Amendment Nos.: 158 & 162

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

Date of initial notice in **Federal Register:** July 20, 1994 (59 FR 37091) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 12, 1994.No significant hazards consideration comments received: No.

Local Public Document Room location: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

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

Date of application for amendments: September 12, 1994

Brief description of amendments: These amendments revise Point Beach Nuclear Plant Technical Specification (TS) 15.3.3, ''Emergency Core Cooling System, Auxiliary Cooling Systems, Air Recirculation Fan Coolers, and Containment Spray,'' by incorporating allowed outage times similar to those contained in NUREG-1431, Revision 0, ''Westinghouse Owner's Group Improved Standard Technical Specifications,'' and by clarifying the operability requirements for the service water pumps. The changes also clarify the completion times for placing a unit in hot or cold shutdown, if a limiting condition for operation cannot be met.

Date of issuance: December 21, 1994

Effective date: Date of issuance, to be implemented within 45 days

Amendment Nos.: 159 & 163

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

Date of initial notice in **Federal Register:** October 24, 1994 (59 FR 53844)The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 21, 1994.No significant hazards consideration comments received: No.

Local Public Document Room location: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241.

Dated at Rockville, Maryland, this 27th day of December 1994.

For The Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[Doc. 95–5 Filed 1–3–95; 8:45 am]

BILLING CODE 7590–01–F

OFFICE OF PERSONNEL MANAGEMENT

Federal Salary Council; Meeting

AGENCY: Office of Personnel Management.

ACTION: Notice of meetings.

SUMMARY: According to the provisions of section 10 of the Federal Advisory Committee Act (P.L. 92–463), notice is hereby given that the forty-second and forty-third meetings of the Federal Salary Council will be held at the time and place shown below. At the meetings the Council will continue discussing issues relating to locality-based comparability payments authorized by the Federal Employees Pay Comparability Act of 1990 (FEPCA). The meetings are open to the public.

DATES: January 30, 1995, at 10:00 a.m.; February 28, 1995, at 10:00 a.m.

ADDRESSES: Office of Personnel Management, 1900 E Street NW., Room 7B09, Washington, DC.

FOR FURTHER INFORMATION CONTACT: Ruth O'Donnell, Chief, Salary Systems Division, Office Of Personnel Management, 1900 E Street NW., Room 6H31, Washington, DC 20415–0001. Telephone number: (202) 606–2838.

For the President's Pay Agent.

Lorraine A. Green,

Deputy Director.

[FR Doc. 95–10 Filed 1–3–95; 8:45 am]

BILLING CODE 6325–01–M

SECURITIES AND EXCHANGE COMMISSION

Under Review by the Office of Management and Budget

Acting Agency Clearance Officer: Richard T. Redfearn, (202) 942–8800.

Upon Written Request Copy Available From: Securities and Exchange Commission, Office of Filings and Information Services, Washington, D.C. 20549.

Reinstatement

Rule 144A Information Request for Qualified Institutional Buyers

[File No. 270–342]

Notice is hereby given that, pursuant to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.) the Securities and Exchange Commission has submitted for reinstatement an information request for issuers regarding market developments under rule 144A. Respondents incur an estimated average burden of 45 minutes to complete the information request.

General comments regarding the estimated burden hours should be directed to the Clearance Officer of the Securities and Exchange Commission at the address below. Any comments concerning the accuracy of the estimated average burden hours for compliance with Commission rules and forms should be directed to Richard T. Redfearn, Acting Director, Office of Information Technology, Securities and Exchange Commission, 450 Fifth Street, N.W., Washington, D.C. 20549 and Clearance Officer for the Securities and Exchange Commission, Office of Management and Budget, (Project No. 3235–0406), New Executive Office Building, Washington, D.C. 20503.

Dated: December 27, 1994.

Margaret H. McFarland,

Deputy Secretary.

[FR Doc. 95–119 Filed 1–3–95; 8:45 am]

BILLING CODE 8010–01–M

[Rel. No. IC–20798; 812–9330]

Dean Witter Select Equity Trust, Select 10 International Series

December 27, 1994.

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

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

APPLICANT: Dean Witter Select Equity Trust, Select 10 International Series.

RELEVANT ACT SECTIONS: Order requested under section 6(c) of the Act that would