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 May 26, 1995, through June 9, 1995. The last biweekly notice was published on Tuesday, June 6, 1995 (60 FR 29869).

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 July 21, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to **(Project Director)**: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

Commonwealth Edison Company, Docket Nos. STN 50-456 and STN 50-457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of amendment request: June 8, 1995, supersedes December 16, 1994, request in its entirety, supplemented by letters dated November 30, 1994, April 27, 1995, May 5 and May 11, 1995.

Description of amendment request: The proposed amendment would revise Figure 3.4-4a in the Braidwood Unit 1's technical specifications which provides the nominal pressurizer power operated relief valve set points for the low-temperature overpressure protection system (LTOPS). The proposed revision would extend the applicability of Figure 3.4-4a from 5.37 effective full power years (EFPY) to 16 EFPY (Unit 1). In addition, the proposed amendment removes the 638 psig administrative limit line from the LTOPS curve, because the appropriate instrument uncertainties and discharge piping pressure limits are included in the proposed LTOPS curve. The amendment request also proposes administrative changes to Figure 3.4-4a format and its associated index page.

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

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

The new LTOPS curve will not change any postulated accident scenarios. The revised curve was developed using industry standards and regulations which are recognized as being inherently conservative. Appropriate instrument uncertainties and allowances have been included in the development of the LTOPS curves. The PT and LTOPS curves provide RCS pressure limits to protect the Reactor Pressure Vessel (RPV) from brittle fracture by clearly separating the region of normal operations from the region where the RPV is subject to brittle fracture.

Using Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, Braidwood Unit 1 Surveillance Capsule U and Capsule X results and the requirements of Appendix G to 10 CFR 50, as modified by the guidance in ASME Code Case N-514, a new LTOPS curve was prepared. This new curve, in conjunction with the PT Limit curves, and the heatup and cooldown ranges provides the required assurance that the RPV is protected from brittle fracture.

No changes to the design of the facility have been made, no new equipment has been installed, and no existing equipment has been removed or modified. This amendment will not change any system operating modes. The revised LTOPS curve provides assurance that the RPV is protected from brittle fracture.

The index page and format changes are purely administrative in nature and are designed to reflect the change in the duration of applicability of Figure 3.4-4a and improve the readability of Figure 3.4-4a. These administrative changes will have no effect on any equipment, system, or operating mode.

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

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

The use of the new LTOPS curve does not change any postulated accident scenarios. The new LTOPS curve was generated using Braidwood capsule surveillance data and an approved, conservative methodology. No new equipment will be installed, and no existing equipment will be modified. No new system interfaces are created, and no existing system interfaces are modified. The new LTOPS curve provides assurance that the RPV is protected from brittle fracture.

No new accident or malfunction mechanism is introduced by this amendment.

The index page and format changes are purely administrative in nature and are designed to reflect the change in the duration of applicability of Figure 3.4-4a, and improve the readability of Figure 3.4-4a. These administrative changes will have no effect on any equipment, system, or operating mode.

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

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

The new LTOPS curve was developed using industry standards and regulations which are recognized as being inherently conservative. Appropriate instrument uncertainties and allowances are included in the development of the new LTOPS curve. This amendment will not change the operational characteristics or design of any equipment or system.

All accident analysis assumptions and conditions will continue to be met. The RPV is adequately protected from non-ductile failure by the revised LTOPS curve.

The index page and format changes are purely administrative in nature and are designed to reflect the change in the duration of applicability of Figure 3.4-4a, and improve the readability of Figure 3.4-4a. These administrative changes will have no effect on any equipment, system, or operating mode.

Thus, the proposed change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481

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

NRC Project Director: Robert A. Capra

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

Date of amendment request: March 4, 1993, as revised April 14, 1993, as supplemented April 19 and May 31, 1995

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to conform to the wording of the revised 10 CFR Part 20, ''Standards for Protection Against Radiation,'' and to reflect a separation of chemistry and radiation protection responsibilities. The supplemental submittals provided additional information on the proposed TS change in response to NRC's request for additional information of May 5, 1995. The original submittal was noticed on May 12, 1993 (58 FR 28053), as corrected June 1, 1993 (58 FR 31222).

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

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

The proposed change does not affect the probability or consequences of an accident. The proposed change is to the ADMINISTRATIVE and RADIOLOGICAL EFFLUENT RELEASES sections of the facility Technical Specifications, and are administrative in nature.

- Change ''Chemistry and Radiation Protection Supervisor'' to ''Radiation Protection Supervisor.''

- The change from ''mR/h'' to ''mrem/h'' is solely a change in terminology since the revised 10 CFR 20 does not recognize or define the roentgen as a unit of radiation.

- The Liquid Effluents Concentration section and the associated bases have been revised to conform with 10 CFR 50.36(a) [10 CFR 50.36a] with effluent concentrations limited to 10 times the limits of 10 CFR 20.1001 - 20.2402, Appendix B, Table 2, Column 2.

- The actual instantaneous dose rate limits of the Gaseous Effluents Dose Rate section have not changed. However, the bases section has. Under the former 10 CFR 20, these dose rates correspond roughly to maximum permissible concentration and dose(s) received by the maximum exposed member of the public if allowed to continue for an entire year. These limits are used more as instantaneous limits (dose rates above which are not allowed to continue for more than one hour at a time) so as to provide assurance not to exceed 10 CFR 50, Appendix I limits.

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

This proposed change is required by the implementation of a new 10 CFR Part 20 requirements (except for the title change) and are administrative in nature (sic). Neither the material condition of the facility nor the accident analyses are affected by this proposed change. Therefore, the proposed change does not create the possibility of a different type of accident than previously evaluated.

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

Each limit that was affected increased the margin of safety by making the limit more conservative; or remained the same.

- The change of distance to ''30 centimeters'' (12 inches) is more conservative, providing a higher degree of protection for occupationally exposed worker.

- The liquid effluent concentration limits remain essentially the same. The bases have changed to [10 CFR 50.36a] reflect 10 times 10 CFR 20.1001 - 20.2402, Appendix B, Table 2, Column 2 limits as controlled by 10 CFR 50.36(a) [10 CFR 50.36a] dose limits.

- Effluent alarm setpoints were reviewed to determine any necessary changes and were found to be set appropriately. No change will be necessary.

- ''The instantaneous release rate limits for airborne releases will not be changed because they are imposed on licensees as a control to ensure that the licensees meet Appendix I requirements.'' Alarm setpoints for these dose rate limits may change slightly due to changes in scientific data and will be reviewed and changed as appropriate prior to implementation.

Therefore, the proposed change 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: North Central Michigan College, 1515 Howard Street, Petoskey, Michigan 49770

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

NRC Project Director: Cynthia A. Carpenter, Acting

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

Date of amendment request: April 12, 1995

Description of amendment request: The amendments delete Technical Specification 3/4.3.4, ''Turbine Overspeed Protection,'' and its associated Bases. The deletion of TS 3/4.3.4 and its associated Bases provides Duke Power Company the flexibility to implement the manufacturer's recommendations for turbine steam valve surveillance test requirements. These test requirements will be relocated from the TS to the Selected Licensee Commitments (SLC) Manual. The SLC Manual is Chapter 16 of the Updated Final Safety Analysis Report.

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

Criterion 1

The requested amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated. Relocation of the affected TS section to the SLC Manual will have no effect on the probability of any accident occurring. In addition, the consequences of an accident will not be impacted since the above system will continue to be utilized in the same manner as before. No impact on the plant response to accidents will be created.

Criterion 2

The requested amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms will be created as a result of relocating the affected TS requirements to the SLC Manual. Plant operation will not be affected by the proposed amendments and no new failure modes will be created.

Criterion 3

The requested amendments will not involve a significant reduction in a margin of safety. No impact upon any plant safety margins will be created. Relocation of the affected TS requirements to the SLC Manual in consistent with the content of the Westinghouse RSTS [Revised Standard Technical Specifications], as the NRC did not require technical specification controls for the turbine overspeed protection system in the RSTS. The proposed amendments are consistent with the NRC philosophy of encouraging utilities to propose amendments that are consistent with the content of the RSTS.

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

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

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

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: May 18, 1995, as supplemented by letter dated May 31, 1995.

Description of amendment request: The proposed amendment would change Tecnical Specification (TS) 3.6.1.2 to defer the next scheduled containment integrated leak rate test (ILRT) at Catawba, Unit 2, for one outage, from the end-of-cycle (EOC) 7 refueling outage (scheduled for October 1995) to EOC-8 (scheduled for March 1997). Title 10 of the Code of Federal Regulations, Part 50, Appendix J, requires that three ILRTs be performed at approximately equal intervals during each 10-year service period at a nuclear station. ''Approximately equal intervals'' is defined in Catawba's TS as 40 plus or minus 10 months. The proposed one-time change would allow Catawba to extend that interval to less than or equal to 70 months.

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

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

Containment leak rate testing is not an initiator of any accident; the proposed interval extension does not affect reactor operations or accident analysis, and has no perceptible radiological consequences. Therefore, this proposed change will not involve a significant increase in the probability or consequences of any previously[]evaluated accident.

2. The proposed change will not create the possibility of any new accident not previously evaluated.

The proposed change does not affect normal plant operations or configuration, nor does it affect leak rate test methods. The test history at Catawba (no ILRT [intergrated leak rate test] failures) provides continued assurance of the leak tightness of the containment structure.

3. There is no significant reduction in a margin of safety.

It has been documented in draft NUREG-1493 that an increase in the ILRT interval from 1 test every 3 years to 1 test every 10 years would result in an increase in population exposure risk in the vicinity of 5 representative plants from .02% to .14%. The proposed change included herein, an increase from 40 [plus or minus] 10 months to [less than or equal to] 70 months, represents a small fraction of that already very small increase in risk. Therefore, it may be concluded that no significant reduction in a margin of safety will occur.

Based on the above, no significant hazards consideration is created by the proposed change.

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

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

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: April 12, 1995

Description of amendment request: The amendments delete Technical Specification 3/4.3.4, ''Turbine Overspeed Protection,'' and its associated Bases. The deletion of TS 3/4.3.4 and its associated Bases provides Duke Power Company the flexibility to implement the manufacturer's recommendations for turbine steam valve surveillance test requirements. These test requirements will be relocated from the TS to the Selected Licensee Commitments (SLC) Manual. The SLC Manual is Chapter 16 of the Updated Final Safety Analysis Report.

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

Criterion 1

The requested amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated. Relocation of the affected TS section to the SLC Manual will have no effect on the probability of any accident occurring. In addition, the consequences of an accident will not be impacted since the above system will continue to be utilized in the same manner as before. No impact on the plant response to accidents will be created.

Criterion 2

The requested amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms will be created as a result of relocating the affected TS requirements to the SLC Manual. Plant operation will not be affected by the proposed amendments and no new failure modes will be created.

Criterion 3

The requested amendments will not involve a significant reduction in a margin of safety. No impact upon any plant safety margins will be created. Relocation of the affected TS requirements to the SLC Manual in consistent with the content of the Westinghouse RSTS [Revised Standard Technical Specifications], as the NRC did not require technical specification controls for the turbine overspeed protection system in the RSTS. The proposed amendments are consistent with the NRC philosophy of encouraging utilities to propose amendments that are consistent with the content of the RSTS.

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

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

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

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: May 17, 1995

Description of amendment request: The amendment will extend the applicability of the current Reactor Coolant System (RCS) Pressure/Temperature Limits and maximum allowed RCS heatup and cooldown rates to 23.6 Effective Full Power Years (EFPY) of operation. In addition, administrative changes are proposed for

TS 3.1.2.1 (Boration Systems Flow Paths-Shutdown) and TS 3.1.2.3 (Charging Pump-Shutdown) to clarify the conditions for which a High Pressure Safety Injection pump may be used.

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:

Pursuant to 10 CFR 50.92, a determination may be made that a proposed license amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed as follows:

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

The pressure-temperature (P/T) limit curves in the Technical Specifications are conservatively generated in accordance with the fracture toughness requirements of 10 CFR 50 Appendix G as supplemented by the ASME Code Section XI, Appendix G recommendations. The RT_{NDT} values are based on Regulatory Guide 1.99, Revision 2, shift prediction and attenuation formula. Analyses of reactor vessel material irradiation surveillance specimens are used to verify the validity of the fluence predictions and the P/T limit curves. Use of these curves in conjunction with the surveillance specimen program ensures that the reactor coolant pressure boundary will behave in a non-brittle manner and that the possibility of rapidly propagating fracture is minimized. Based on the use of plant specific material data, analysis has demonstrated that the current P/T limit curves will remain conservative for up to 23.6 EFPY.

In conjunction with extending the applicability of the existing P/T limit curves, the low temperature overpressure protection (LTOP) analysis for 15 EFPY is also extended. The LTOP analysis confirms that the current setpoints for the power-operated relief valves (PORVs) will provide the appropriate overpressure protection at low Reactor Coolant System (RCS) temperatures. Because the P/T limit curves have not changed, the existing LTOP values have not changed, which include the PORV setpoints, heatup and cooldown rates, and disabling of non-essential components.

The proposed amendment does not change the configuration or operation of the plant, and assurance is provided that reactor vessel integrity will be maintained. Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

By applying plant specific data in the determination of critical vessel material limits, the applicability of the existing pressure temperature limits and LTOP requirements can be extended. There is no change in the configuration or operation of the facility as a result of the proposed amendment. The amendment does not involve the addition of new equipment or the modification of existing equipment, nor does it alter the design of St. Lucie plant systems. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Analysis has demonstrated that the fracture toughness requirements of 10 CFR 50 Appendix G are satisfied and that conservative operating restrictions are maintained for the purpose of low temperature overpressure protection. The P/T limit curves will provide assurance that the RCS pressure boundary will behave in a ductile manner and that the probability of a rapidly propagating fracture is minimized. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Based on the discussion presented above and on the supporting Evaluation of Proposed TS Changes, FPL has concluded that this proposed license amendment involves no significant hazards consideration.

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

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

Attorney for licensee: J. R. Newman, Esquire, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Project Director: David B. Matthews

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

Date of amendment request: May 17, 1995

Description of amendment request: The proposed amendments will improve consistency between the Technical Specifications and the improved Combustion Engineering Standard Technical Specifications (NUREG-1432, dated September 1992) by incorporating changes in text and resolving other inconsistencies identified by the NRC and plant operations staff.

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:

Pursuant to 10 CFR 50.92, a determination may be made that a proposed license amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed as follows:

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

The proposed amendments consist of administrative changes to the Technical Specifications (TS) for St. Lucie Units 1 and 2. The amendments will implement changes in text to improve consistency within the TS for each unit, the improved Combustion Engineering Standard Technical Specifications (NUREG-1432, dated September 1992), and the regulations. The proposed amendments do not involve changes to the configuration or method of operation of plant equipment that is used to mitigate the consequences of an accident, nor do the changes otherwise affect the initial conditions or conservatism assumed in any of the plant accident analyses. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed administrative revisions will not change the physical plant or the modes of plant operation defined in the Facility License for each unit. The changes do not involve the addition or modification of equipment nor do they alter the design or operation of plant systems. Therefore, operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not

involve a significant reduction in a margin of safety.

The proposed amendments are administrative in nature and do not change the basis for any technical specification that is related to the establishment of, or the preservation of, a nuclear safety margin. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

Based on the above discussion and the supporting Evaluation of Technical Specification changes, FPL has determined that the proposed license amendment involves no significant hazards consideration.

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

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

Attorney for licensee: J.R. Newman, Esquire, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Project Director: David B. Matthews

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

Date of amendment request: May 23, 1995

Description of amendment request: The licensee proposes to change Turkey Point Units 3 and 4 Technical Specifications (TS) by changing the setpoint presentation format for the Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) instrumentation setpoints contained in Technical Specification Tables 2.2-1 and 3.3-3. The approved Westinghouse five-column instrument setpoint methodology currently being used to establishing those setpoints would be retained. The intent of the amendments is to eliminate the need for minor administrative license amendments to these tables that do not impact either the Trip Setpoints or the Safety Analysis Limits.

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

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

No changes to the Reactor Trip System instrumentation setpoints, ESFAS instrumentation setpoints, or the Turkey Point Plant licensing basis (NRC-approved, Westinghouse five-column setpoint methodology, as documented in Westinghouse topical report WCAP-12745P), is being made. The changes proposed reduce the level of detail in the Technical Specifications and place that detailed information in controlled procedures, drawings and the Final Safety Analysis Report. Since the setpoints and methodology remain the same, the changes proposed by this submittal will not increase the probability or consequences of an accident previously evaluated.

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

These proposed changes remove from the Technical Specifications a level of detail which will be maintained in controlled procedures and drawings. The Turkey Point Plant licensing basis (NRC-approved, Westinghouse five column setpoint methodology, as documented in Westinghouse topical report WCAP-12745P), continues to be used to calculate the Reactor Trip System and ESFAS setpoints. No changes to Reactor Trip System or ESFAS instrumentation setpoints are proposed. Since the same methodology will be used to determine the setpoints and no setpoints are changed, the possibility that a new or different kind of accident from any previously evaluated will not be created.

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

The Turkey Point Plant licensing basis (NRC-approved, Westinghouse five column setpoint methodology, as documented in Westinghouse topical report WCAP-12745P), continues to be used to calculate the Reactor Trip System and ESFAS setpoints. No changes to the Reactor Trip System or ESFAS instrumentation setpoints are proposed. Since the same methodology will be used to determine the setpoints, and no setpoints are changed by this submittal, this change does not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Florida International University, University Park, Miami, Florida 33199

Attorney for licensee: J.R. Newman, Esquire, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Project Director: David B. Matthews

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: June 6, 1995

Description of amendment request: The proposed change would revise Plant Hatch Units 1 and 2 Technical Specification (TS) Surveillance Requirements (SR) 3.6.4.1.3 and 3.6.4.1.4 for the secondary containment drawdown. The revision would reduce the SR acceptance criteria to greater than or equal to 0.20 inch of vacuum from greater than or equal to 0.25 inch of vacuum. Also, the licensee proposed to change the Bases to reflect the proposed TS revision.

The licensee stated that the secondary containment performs no active function in response to either loss-of-coolant accident or fuel handling accident. However, its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped within the secondary containment structure will be treated by the Unit 1 and Unit 2 standby gas treatment systems prior to discharge to the environment. This change will continue to provide adequate margin for the secondary containment to be sufficiently leak tight such that the conclusions of the accident analysis remain valid.

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 change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The secondary containment serves a mitigation function and therefore this change does not increase the probability of an accident previously evaluated. The consequences of the previously evaluated accidents are not affected because at the wind conditions assumed in the accident analysis the building will be at a negative pressure and no exfiltration is postulated. Furthermore, the estimated wind speed at which exfiltration might take place (31 mph) is not a frequent occurrence (wind speeds of greater than 24 mph occur [less than] <0.5% of the time based on Plant Hatch specific meteorological data).

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

requirement acceptance criteria does not physically modify the plant nor does it modify the operation of any existing equipment.

3. The proposed change does not involve a significant reduction in the margin of safety. The change in vacuum acceptance criteria results in a slightly lower wind speed that may result in exfiltration from the building. However, this wind speed (31 mph) is in the realm of wind speeds which are infrequent at Plant Hatch. Furthermore, there are numerous conservatisms in the existing dose calculations including: neutral to stable meteorological conditions, ground level release until establishment of the required vacuum, accident source terms at event initiation, and no credit for plateout. The secondary containment would be maintained at a slight negative pressure shortly after the Standby Gas Treatment fans are running and the releases would be from the main stack (well before the accident source term would be present in the secondary containment). Some plateout would also occur and this is conservatively ignored. Therefore the margin of safety 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 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

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

Date of amendment request: May 17, 1995

Description of amendment request: The proposed license amendment would revise Section 3.2 of the Technical Specifications (TSs) for Three Mile Island Nuclear Station, Unit 1 (TMI-1) to relocate the requirements for volume and boron concentration of the chemical addition system boric acid mix tank and the reclaimed boric acid storage tank from the TMI-1 TSs to the TMI-1 Core Operating Limits Report. The licensee, in its request, stated that the proposed changes are consistent with the intent of NRC Generic Letter 88-16.

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

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated. The proposed amendment relocates chemical addition tank volume and boron concentration parameters from Technical Specifications to the TMI-1 Core Operating Limits Report. The proposed amendment provides continued control of the values of these parameters and assures these values are developed using NRC-approved reload methodologies consistent with all applicable limits of the safety analyses addressed in the TMI-1 [Final Safety Analysis Report] FSAR. The Technical Specifications retain the requirement to maintain the plant within the appropriate bounds of these limits. Therefore, the proposed amendment has no effect on the probability of occurrence or consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment relocates chemical addition tank volume and boron concentration parameters to the TMI-1 Core Operating Limits Report. The Technical Specifications retain the requirement to maintain the boric acid mix tank and reclaimed boric acid storage tank volume and boron concentration parameters within the appropriate limits. Therefore, the proposed amendment has no effect on the possibility of creating a new or different kind of accident from any accident previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. The proposed amendment provides continued control of the boric acid mix tank and reclaimed boric acid storage tank volume and boron concentration parameters and assures these values remain consistent with all applicable limits of the safety analyses addressed in the TMI-1 FSAR. Therefore, it is concluded that operation of the facility in accordance with 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: Law/Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

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

NRC Project Director: Phillip F. McKee

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

Date of amendment request: May 24, 1995

Description of amendment request: The proposed license amendment would revise Table 4.1-1 of the Technical Specifications (TSs) for Three Mile Island Nuclear Station, Unit 1 (TMI-1) to revise the test frequency requirement for the source range nuclear instrumentation from 7 days before reactor startup to 6 months before startup.

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

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

The proposed revision to the Technical Specifications does not involve any physical changes to the plant, and it does not impact the safety analysis with respect to design basis events and assumptions. The only change proposed is in the ''Test'' frequency for source-range Nuclear Instrumentation by revision of the appropriate Tech. Spec. tables. The revised testing requirement has no impact upon the probability of occurrence or the consequences of any accident previously evaluated, because no credit is taken in the accident analyses for the source range monitors nor are there any inputs to the Reactor Protection System. Tech. Spec. 3.1.9.2 requires that the control rod withdraw inhibit be operable at all times; however, it is not affected by this change request. Additionally, no nuclear safety equipment or systems interface with source-range nuclear instrumentation, and operator ability to monitor and trend post-accident neutron level is not affected by the proposed change. Therefore, this change request will not increase the probability of occurrence or the consequences of any previously analyzed accidents as described in the Updated [Final Safety Analysis Report] FSAR (UFSAR).

2. Operation of the facility in accordance with the proposed TSCR would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed revision to the TMI-1 Technical Specifications does not involve any physical changes to the plant, and does not impact on the safety analysis with respect to design basis events and assumptions. The only change proposed is in the ''Test'' frequency for Nuclear Instrumentation by revision of the appropriate Tech. Spec. tables. No nuclear safety equipment or systems interface with the source-range nuclear instrumentation, and operator ability to monitor and trend post-accident neutron levels is not adversely affected by the proposed change. In addition, the source-range nuclear instrument channels provide indication to the control room, plant computer and one of two channels provides input to Remote Shutdown Panel B.

The 0.5% instrument drift over a six (6) month period will not affect the ability to operate other safety equipment; nor, will it increase the probability of failure of the rod withdrawal inhibit. The inhibit function is triggered by a startup rate, and a 0.5% drift over six (6) months will not affect the instrument's ability to perform the inhibit function. Therefore, this change has no impact upon the possibility of creating a new or different kind of accident from any previously evaluated in the UFSAR.

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

The proposed revision to the TMI-1 Technical Specifications does not involve any physical changes to the plant, and does not impact on the safety analysis with respect to design basis events and assumptions. The only change proposed is in the surveillance frequency for Nuclear Instrumentation by revision of the appropriate Tech. Spec. tables. Startup rate instrumentation is not included in Technical Specifications 2.0, ''Safety Limits''; and, hence, all system Limiting Conditions for Operation(s) remain unchanged. Testing of the source-range nuclear instrument channels within six (6) months prior to a reactor startup will not decrease the margin of safety. Hence, the margin of safety for the plant is not diminished by this change request.

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: Law/Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

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

NRC Project Director: Phillip F. McKee

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

Date of amendment request: June 1, 1995

Description of amendment request: The proposed license amendment would revise Section 5.3.1.1 of the Technical Specifications (TSs) for Three Mile Island Nuclear Station, Unit 1 (TMI-1) to allow use of an alternate zirconium-based cladding material manufactured by Babcock & Wilcox Fuel Company to test the properties of the fuel in an operating core. Present TSs require fuel clad material to be either ''zircaloy'' or ''ZIRLO.''

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

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. The test assemblies with the zirconium-based claddings are mechanically and thermal-hydraulically similar to the remainder of the reload batch and the rest of the core, so no failure probability is increased, nor is any operational practice changed which could introduce a new initiator of an accident. The only credible event which could occur as a result of this demonstration is clad failure of the test fuel rods. The number of fuel rods involved is such a small percentage of the core inventory that even a postulated failure of all the demonstration fuel rods from a cause related to the demonstration would not result in dose consequences greater than existing limits. A failure of the fuel rods from a cause not related to the demonstration would not result in consequences greater than those which would have occurred had the assemblies not been demonstrated assemblies. Therefore, this change does not increase the probability of occurrence or the consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. The mechanical and thermal-hydraulic similarity of the test assemblies to the remainder of assemblies in the core precludes the credible possibility of creating any new failure mode or accident sequence. The use of the demonstration assemblies does not involve any alterations to plant equipment or procedures which would introduce any new or unique operational modes or accident precursors.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. The demonstration assemblies meet the same design as the remainder of assemblies in the core. Existing reload design and safety analysis limits are maintained, and the FSAR analyses are bounding. No special setpoints or other safety settings are required as a result of the use of these two (2) test assemblies. The assemblies will be placed in locations which will not experience limiting peak power conditions. Therefore, it is concluded that operation of the facility in accordance with the proposed amendment does not involve a reduction in a margin of safety.

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

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

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

NRC Project Director: Phillip F. McKee

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

Date of amendment request: April 27, 1995, as supplemented by letters dated May 4, and May 25, 1995.

Description of amendment request: The proposed amendment would change the tables associated with Technical Specification (TS) 3/4.3.3.5, Remote Shutdown System, to eliminate the core exit thermocouples (CETs). The proposed amendment would also change the tables associated with TS 3/4.3.3.6, Accident Monitoring Instrumentation, to require two operable channels of CETs, where each channel would be required to have at least two operable CETs per core quadrant. Each channel would also be required to have at least four operable CETs in at least one quadrant to support the operability of the subcooling margin monitors. In addition, the actions related to TS 3/4.3.3.6 would be changed to require that a report be submitted if one CET channel in a quadrant is inoperable for more than 30 days, and require a plant shutdown if both CET channels in a quadrant are inoperable for more than 7 days.

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

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

Change to Technical Specification 3.3.3.5:

Deleting the reference to the core exit thermocouples from the Remote Shutdown Technical Specification will not involve a significant increase in the probability of an accident previously evaluated because the core exit thermocouples are not potential accident initiators. The consequences of an accident previously evaluated will not be increased because the core exit thermocouples availability is not reduced, since adequate assurance of their operability is provided in Technical Specification 3.3.3.6, and by the surveillance of other indications that require the availability of the displays that also provide the core exit temperatures at the Auxiliary Shutdown Panel.

Change to Technical Specification 3.3.3.6: The proposed change reduces the number of core exit thermocouples required per quadrant per channel from at least 4 to at least 2. Thus, the Actions when less than 4 thermocouples per quadrant per train are Operable but more than 6 thermocouples per quadrant are OPERABLE, and less than 6 thermocouples per quadrant are OPERABLE but at least 4 thermocouples per quadrant are OPERABLE and with the number of OPERABLE channels less than 4 thermocouples per quadrant are being deleted. This change does not affect the probability of an accident. The Accident Monitoring Instruments are not initiators of any analyzed events. The consequence of an accident is not affected by this change. The requirement to have two core exit thermocouples OPERABLE per quadrant per channel is adequate because one OPERABLE core exit thermocouple must be located near the center of the core and the other OPERABLE core exit thermocouple must be located near the core perimeter, such that the pair of core exit thermocouples indicate the radial temperature gradient across their core quadrant. The change will not alter assumptions relative to the mitigation of an accident or transient event. Functions supported by the thermocouples will still be adequately supported by the system. The revised specification provides for at least one quadrant per channel to have at least four operable thermocouples to protect the subcooling margin monitor in the event of a single failure. The other indications used to assess core cooling, as described in Chapter 7B of the South Texas Project Updated Final Safety Analysis Report remain unaffected by the proposed change. Therefore, this change will not involve a significant increase in the probability or consequence of an accident previously evaluated.

The proposed change also affects the allowed outage times for the thermocouples. The existing specification allows for 31 days in the case where there are less than four thermocouples per quadrant per train operable, 7 days where there are less than 6 thermocouples per quadrant, and 48 hours where there are less than 4 thermocouples per quadrant. The required action for each of these cases is a plant shutdown. The proposed specification will require a report to the Commission after 30 days in the case where one channel of core exit thermocouples is inoperable, and it will require the plant to go to HOT SHUTDOWN if two channels are inoperable for more than 7 days. A plant shutdown with only one channel inoperable is not warranted based on the fact that the redundant channel remains available to provide the necessary indication and the passive nature of the instrumentation (i.e., no critical automatic action).

As noted above, the core exit thermocouples are not accident initiators; consequently, the change in allowed outage time does not affect the probability of an accident. The consequences of an accident are not significantly increased because the changes to the allowed outage times are not extended to allow operation of the system in such a degraded condition that it will not perform its function. In addition, the other indications used to assess core cooling, as described in Chapter 7B of the South Texas Project Updated Final Safety Analysis Report remain unaffected by the proposed change. As noted above, functionality of the core exit temperature indication is preserved by requiring at least two thermocouples to be operable in separate regions of the core quadrant.

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

Change to Technical Specification 3.3.3.5: Deleting the core exit thermocouples from the Remote Shutdown Technical Specification will not create the possibility of a new or different accident because there are no automatic actuations performed by the core exit thermocouples, nor are any different plant configurations or different operational procedures proposed. The existing safety analyses are unchanged and still applicable.

Change to Technical Specification 3.3.3.6: The proposed change reduces the number of core exit thermocouples required per quadrant per channel from at least 4 to at least 2. Thus, the Actions when less than 4 thermocouples per quadrant per train are Operable but more than 6 thermocouples per quadrant are OPERABLE, and less than 6 thermocouples per quadrant are OPERABLE but at least 4 thermocouples per quadrant are OPERABLE and with the number of OPERABLE channels less than 4 thermocouples per quadrant are being deleted. This change will not physically alter the plant (no new or different type of equipment will be installed). The changes in methods governing normal plant operation are consistent with current safety analysis assumptions. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The change in the allowed outage time does not alter the physical configuration of the plant or how the plant is operated; consequently, this change does not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

Change to Technical Specification 3.3.3.5: Deleting the core exit thermocouples from the Remote Shutdown Technical Specification does not involve a significant reduction in the margin of safety because the core exit thermocouples indications will still be available at the Auxiliary Shutdown Panel. In addition, adequate and appropriate assurance of the operability of the core exit thermocouples is provided in Technical Specification 3.3.3.6 for Accident Monitoring Instrumentation, including the changes proposed in this letter.

Change to Technical Specification 3.3.3.6: The proposed change reduces the number of core exit thermocouples required per quadrant per channel from at least 4 to at least 2. Thus, the Actions when less than 4 thermocouples per quadrant per train are Operable but more than 6 thermocouples per quadrant are OPERABLE, and less than 6 thermocouples per quadrant are OPERABLE but at least 4 thermocouples per quadrant are OPERABLE and with the number of OPERABLE channels less than 4 thermocouples per quadrant are being deleted. The margin of safety is not affected by this change. The Accident Monitoring Instrumentation provide no automatic actuation functions. Even though the number of core exit thermocouples per quadrant per channel is being reduced, the Bases requirement to have one core exit thermocouple located near the center of the core and one core exit thermocouple located near the core perimeter ensures that the pair of core exit thermocouples indicate the radial temperature gradient across their core quadrant which ensures the required level of information is available. The functions dependent on the core exit thermocouples are still adequately supported by the thermocouples. The revised specification provides for at least one quadrant per channel to have at least four operable thermocouples to protect the subcooling margin monitor in the event of a single failure. In addition, the other indications used to assess core cooling, as described in Chapter 7B of the South Texas Project Updated Final Safety Analysis Report remain unaffected by the proposed change. The safety analysis assumptions will still be maintained, thus, no question of safety exists. Therefore, the change does not involve a significant reduction in a margin of safety.

The proposed changes to the allowed outage times have no significant impact on the margin of safety. A plant shutdown with only one channel inoperable is not warranted based on the fact that the redundant channel remains available to provide the necessary indication and the passive nature of the instrumentation (i.e., no critical automatic action). Based on the small likelihood of an accident occurring concurrent with the station being in an ACTION statement with regard to the thermocouples, and the small chance that the degradation of the system in such a situation would affect its functionality, and the diversity provided by other indications of core cooling, the changes in the allowed outage times are not considered significant.

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

Local Public Document Room location: Wharton County Junior

College, J. M. Hodges, Learning Center, 911 Boling Highway, Wharton, Texas 77488

Attorney for licensee: Jack R. Newman, Esq., Newman & Holtzinger, P.C., 1615 L Street, N.W., Washington, D.C. 20036

NRC Project Director: William D. Beckner

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

Date of amendment requests: March 31, 1995

Description of amendment requests: The proposed amendments would modify the technical specifications to eliminate the requirement to test certain safeguards pumps via their recirculation flowpath. The affected pumps are the centrifugal charging pumps, residual heat removal pumps, motor driven auxiliary feedwater pumps, and the turbine driven auxiliary feedwater pumps. The proposed amendments would also eliminate references to specific discharge pressures and flows associated with these pumps and remove footnotes associated with the Unit 2 cycle 9-10 refueling outage which are no longer applicable.

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

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

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

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

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

Criterion 1

The purpose for conducting periodic testing of the pumps identified in this proposed amendment is to detect gross degradation as required by Section XI of the ASME [American Society of Mechanical Engineers] Code. The Cook Nuclear Plant IST [Inservice Testing] program, which encompasses Section XI of the ASME Code, is the basis for the existing as well as the proposed T/Ss. Testing the pumps utilizing a high capacity flowpath instead of a recirculation flow path (where applicable) will have no impact on the ability of the pump to perform its intended function. In fact, it is expected that the high capacity flowpath will provide a more accurate assessment of the pump/systems' conditions and ability to meet their safety function.

The removal of specific test parameters, in favor of referencing the Cook Nuclear Plant IST Program, will not impact the ability of the pumps to perform their safety related function. IST Program parameters ensure that the pumps under test provide the support assumed in the plant's safety analyses.

Therefore, based on these considerations, it is concluded that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2

The proposed change will preclude the need to realign selected pumps to their recirculation flowpaths for testing purposes (where applicable). Eliminating the need for alignment to the recirculation flowpath aids in maximizing the pump's availability to perform its safety function.

As stated previously the removal of the specific test parameters, in favor of referencing the Cook Nuclear Plant IST Program will not impact the ability of the pumps to perform their intended safety function.

Thus, it is concluded that the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3

As stated previously, testing of the selected pumps utilizing a high capacity flowpath will provide greater assurance of pump capability and maximize pump availability. Additionally, removing specific test parameters in favor of referencing the Cook Nuclear Plant IST Program will have no impact on the ability of the pumps to perform their intended safety function. Therefore, we believe that the margin for safety as defined int 10 CFR [Part] 100 has not been reduced. Based on these considerations, it is concluded that the changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Although not specifically addressed in the licensee's analysis, the elimination of specific discharge pressures and flows is encompassed in the elimination of the recirculation testing requirement and presents no additional significant hazards consideration. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

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

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

NRC Project Director: Cynthia A. Carpenter, Acting

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

Date of amendment requests: May 19, 1995

Description of amendment requests: The proposed amendments would modify the Technical Specification action statement associated with the Main Steam Safety Valves (MSSVs). The action statement would reflect different requirements based on operating Mode and the power range neutron flux high setpoint with inoperable MSSVs would be revised in response to an issue raised in Westinghouse Nuclear Safety Advisory Letter 94-001.

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

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

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

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

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

Criterion 1

Correction of the setpoint methodology does not represent a credible accident initiator. The new methodology reduces the allowable power level setpoints and is conservative compared to the presently evaluated setpoints. The consequences of any previously evaluated accident are not adversely affected by this action because the decrease in the setpoints resulting from the new calculational methodology will ensure that the MSSVs are capable of relieving the pressure at the allowable power levels. Based on these considerations, it is concluded that the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Correcting the overly restrictive action statements of T/S 3.7.1 does not involve a significant increase in the probability of an accident. The proposed changes modify existing text to more accurately reflect the intention of the restrictions imposed by the action statements. The changes do not create any situation that would initiate a credible accident sequence.

Criterion 2

The change in Table 3.7-1 reduces the allowable power levels that can be achieved in the event that one or more main steam safety valve(s) is inoperable. This change is a result of vendor guidance to correct an error in the existing methodology used to determine the setpoints for the power level. Changing the methodology used to determine the setpoints, and lowering the setpoints themselves, do no create a new condition that could lead to a credible accident. Therefore, it is concluded that the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The action statements remain in effect to perform the intended function of protecting the plant's secondary side when the main steam safety valves are inoperable. They have only been modified to correct the overly restrictive language that specifies when, in each MODE, specific actions must be taken. Therefore, the proposed change does not create a new or different type of accident.

Criterion 3

The margin of safety presently provided is not reduced by the proposed change in the setpoints. The change will correct the limiting power levels that are to be implemented when MSSVs are inoperable. This action does not adversely affect the margin that was previously allocated for the ability of the MSSVs to relieve secondary side pressure. Based on these considerations, it is concluded that the changes do not involve a significant reduction in a margin of safety.

The margin of safety is also not significantly reduced by the proposed change to the action statements of the T/S. The proposed revision clarifies when specific actions are to be taken in response to inoperable main steam safety valves. The changes do not decrease the effectiveness of the actions to be taken; therefore, they do not significantly reduce any margin of safety.

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

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

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

NRC Project Director: Cynthia A. Carpenter, Acting

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

Date of amendment request: April 16, 1995

Description of amendment request: The proposed amendment would modify certain requirements of the Seabrook Station Technical Specifications relating to containment building penetrations during refueling operations. One change would allow both doors of the containment personnel airlock (PAL) to be open during core alterations or movement of irradiated fuel within containment provided at least one PAL door is capable of being closed and a designated individual is available outside the PAL to close the door. Another change would allow the use of alternate containment building penetration closure methodologies during refueling operations and provide for the manual closure of a penetration provided a designated individual is available at the penetration.

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

A. The changes do not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)). The changes do not affect the events or conditions which could result in a fuel handling accident and do not affect any equipment or procedures used for fuel handling. The changes would continue to ensure that penetrations which provide direct access of the containment atmosphere to outside containment are capable of restricting a release of radioactive material to the environment. Therefore, the changes do not involve a significant increase in the probability of an accident previously evaluated.

The changes do have the potential for increased dose at the site boundary due to a postulated fuel handling accident. However, the licensee's radiological evaluations show that the resulting offsite and control room doses would be well within the acceptance limits of 10 CFR Part 100 and within the acceptance limits of GDC 19.

The Commission has provided guidance concerning the application of standards in 10 CFR 50.92 by providing certain examples (cf. FEDERAL REGISTER, March 6, 1986 51 FR 7751) of amendments that are considered not likely to involve a significant hazards consideration. These changes are similar to example (vi) in the **Federal Register** notice, in that they result in an increase in the consequences of a previously analyzed accident, but the results of the change are clearly within all acceptance criteria.

B. The changes do not create the possibility of a new or different kindof accident from any accident previously evaluated (10 CFR 50.92(c)(2))because the changes do not affect the events or conditions which could result in a fuel handling accident and do not affect any equipment or procedures used for fuel handling. The changes do not make any modifications to existing plant structures, systems, or components, or otherwise affect the manner by which the facility is operated.

C. The changes do not involve a significant reduction in a margin ofsafety (10 CFR 50.92(c)(3)) because the increase in calculated offsite and control room doses resulting from a postulated fuel handling accident are within the acceptance limits of 10 CFR Part 100 and within the acceptance limits of GDC 19. Additionally, the changes do not otherwise affect the manner by which the facility is operated or involve modifications to equipment or features which affect the operational characteristics of the facility.

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

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

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

NRC Project Director: Phillip F. McKee

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

Date of amendment request: May 18, 1995

Description of amendment request: The proposed amendment revises the minimum temperature at which the reactor vessel head bolting studs are allowed to be placed under tension. In addition, the proposed amendment revises the minimum reactor vessel metal temperature during core critical operation, revises the minimum reactor vessel metal temperature for pressure tests, makes editorial changes, and revises the bases for the applicable section.

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

NNECO has reviewed the proposed changes against the criteria set forth in 10CFR50.92 and has concluded that the changes do not involve a significant hazards consideration (SHC). The bases for this conclusion are that the three criteria of 10CFR50.92(c), discussed separately below, are not compromised. The proposed changes do not involve a SHC because the changes would not:

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

Revising the boltup temperature of the reactor vessel head, from 86—F to 70—F, does not decrease the margins of safety, as required by 10CFR 50 Appendix G, against non-ductile failure of the reactor vessel. Therefore, the probability of occurrence of an accident previously evaluated in the safety analysis report (i.e., a LOCA)[loss of coolant accident] is not increased since the revised boltup temperature does not increase the probability of failure of the vessel head flange region. The reactor vessel is a passive component which does not initiate or play a role in any previously evaluated accidents or in mitigating the consequences of any previously evaluated accidents. Therefore, the proposed changes do not involve a significant increase in the probability or the consequences of any accident previously evaluated.

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

Revising the boltup temperature of the reactor vessel head, from 86°F to 70°F, does not decrease the margins of safety, as required by 10CFR 50 Appendix G, against non-ductile failure of the reactor vessel. Therefore, the possibility for a new or different kind of accident than previously evaluated (i.e., a LOCA through the vessel flange) is not created.

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

Using the proposed boltup temperature of 70°F still provides a self-imposed ''margin'' over the most limiting vessel flange region RT_{NDT} of 22°F (i.e., 70° - 48° = 22°). This is a ''margin'' over and above the boltup temperature required by Appendix G to the 1992 ASME Section XI Code, since Appendix G would allow a boltup temperature of 48—F.

The above proposed changes to the Limiting Condition for Operation for tensioning the reactor vessel head studs do not alter the configuration, normal operation, design bases, function, mission, or performance of the subject components. Therefore, the proposed changes do not affect the margin of safety inherent in the design, analysis, function, or operation of the reactor vessel head flange region. The proposed changes do not alter the fuel clad barrier, fuel integrity, reactor vessel integrity, reactor coolant system integrity, or the containment boundary integrity; thus the margin of safety related to these barriers remains unchanged.

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

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

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

NRC Project Director: Phillip F. McKee

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

Date of amendment request: May 24, 1995

Description of amendment request: The proposed amendment would permit an individual who does not have a current senior reactor operator (SRO) license to hold the Operations Manager position. The position will require the individual to have previously held an SRO license at a boiling water reactor (BWR). An individual serving in the capacity of the Assistant Operations Manager will hold a current SRO license for Millstone Unit 1, if the Operations Manager does not. In addition, the proposed amendment would renumber the applicable sections of the related 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:

NNECO has reviewed the proposed change in accordance with 10CFR50.92 and concluded that the change does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed change does not involve an SHC because the changes would not:

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

The proposed change affects an administrative control, which was based on the guidance of ANSI N18.1-1971. ANSI N18.1-1971 recommended that the Operations Manager hold an SRO license. The current guidance in Section 4.2.2 of ANSI/ANS 3.1-1987 recommends, as one option, that the Operations Manager have held a license for a similar unit and the Operations Middle Manager hold an SRO license. While the Operations Middle Manager position does not exist at Millstone Unit No. 1, NNECO has created the position of Assistant Operations Manager. The individual in this position would meet the requirements for, and would have responsibilities as recommended in, ANSI/ANS 3.1-1987 for the Operations Middle Manager position.

Therefore, the proposed change requests an exception to ANSI N18.1-1971 to allow use of ANSI/ANS 3.1-1987 in a limited circumstance. Specifically, the proposed revision to Technical Specification 6.3.1 would require the Operations Manager to either hold an SRO license at Millstone Unit No. 1 or have held an SRO at a BWR.

If the Operations Manager does not hold an SRO license at Millstone Unit No. 1, the specification will require the Assistant Operations Manager to hold, and continue to hold, an SRO license. The proposed change includes the requirement to have held a license for a similar unit (a BWR) in accordance with Section 4.2.2 of ANSI/ANS 3.1-1987, if the Operations Manager does not hold an SRO license at Millstone Unit No. 1. For those areas of knowledge that require an SRO license, the Assistant Operations Manager will provide the technical guidance typically provided by the Operations Manager.

The proposed change does not alter the design of any system, structure, or component, nor does it change the way plant systems are operated. It does not reduce the knowledge, qualifications, or skills of licensed operators, and does not affect the way the Operations Department is managed by the Operations Manager. The Operations Manager will continue to maintain the effective performance of his personnel and ensure the plant is operated safely and in accordance with the requirements of the operating license. Additionally, the control room operators will continue to be supervised by the licensed Shift Supervisor.

The proposed change does not detract from the Operations Manager's ability to perform his primary responsibilities. In this case, by having previously held an SRO license, the Operations Manager has achieved the necessary training, skills, and experience to fully understand the operation of plant equipment and the watch requirements for operators. In summary, the proposed change does not affect the ability of the Operations Manager to provide the plant oversight required of that position. Thus, it does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to Technical Specification 6.3.1 does not affect the design or function of any plant system, structure, or component, nor does it change the way plant systems are operated. It does not affect the performance of licensed operators. Operation of the plant in conformance with technical specifications and other license requirements will continue to be supervised by personnel who hold an SRO license. The proposed change to Technical Specification 6.3.1 ensures that the Operation Manager will be a knowledgeable and qualified individual by requiring the individual to have held an SRO license at a BWR. Based on the above, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change involves an administrative control that is not related to the margin of safety. The proposed change does not reduce the level of knowledge or experience required of an individual who fills the Operations Manager position, nor does it affect the conservative manner in which the plant is operated. The Control Room operators will continue to be supervised by personnel who hold an SRO license. Thus, the proposed change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Phillip F. McKee

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

Date of amendment request: May 26, 1995

Description of amendment request: The proposed amendment will delete the old limiting conditions for operation (LCOs) and surveillance requirements and add new LCOs, surveillance requirements, and bases for the loss of normal power (LNP) instrumentation system.

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

NNECO has reviewed this proposed change in accordance with 10CFR50.92 and concluded that this change does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed change does not involve an SHC because the changes would not:

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

The change does not increase the probability of a loss of off-site power event or the occurrence of any accidents which assume loss of off-site power. This is ensured by the LNP instrumentation system design which uses multiple sensing relays, redundancy, and qualified Class 1E components, as well as conservative operability and surveillance requirements.

Full LNP logic requires two sets of relays to trip in one of two redundant groups. One set monitors bus 14E and the other set monitors bus 14F. Separate sets are provided for loss of voltage and degraded voltage monitoring. This design minimizes the likelihood of an inadvertent full LNP initiation. To maintain redundancy in the instrumentation, two separate groups are provided, each group being powered from an independent DC supply. Partial LNP logic is also provided to detect a loss of voltage on a single emergency bus. Redundancy in the partial LNP logic is achieved by providing an independent logic for each emergency power train.

The proposed technical specification would require that the LNP instrumentation be maintained operable except when the unit is in cold shutdown or refueling conditions. If redundancy in the ability to detect a loss of voltage or degraded voltage and initiate a full LNP is not maintained, reactor operation would be permitted for seven days. In this situation, both full and partial LNP (and both emergency power sources) remain operable. An action statement of seven days, which is the same as the action statement duration for an inoperable EDG [emergency diesel generator], is justified based on continued operability of the other LNP group. Additionally, it allows a reasonable amount of time to perform repairs.

The time delays and voltage setpoints specified in Table 3.2.4 ensure that the emergency power source starting and loading times continue to meet the current technical specification requirements. Also, these time delays are long enough to preclude false trips due to anticipated voltage transients (e.g., during motor starts). The relay calibration surveillance procedure will establish acceptance criteria for each relay to ensure that the total times specified in Table 3.2.4 are not exceeded. The proposed surveillance testing and calibration frequency of every refueling outage is consistent with the requirements in the current technical specification.

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

There are no new failure modes associated with this change since the proposed requirements will ensure the LNP instrumentation system is available to perform its safety function. Individual voltage sensing relays, when removed from their cases, would provide the tripped contact configuration. The proposed technical specification would allow relays to be placed in the tripped condition as long as it would not inhibit the LNP function or cause an inadvertent initiation. Additionally, since the design function to ensure that adequate power is available to operate the emergency safeguards equipment has not changed, no new accident or accident of a different kind is created.

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

The protective boundaries are not affected because the consequences of any design basis accident are not changed. Since the protective boundaries are not affected, the safety limits are also unaffected. The proposed change maintains the basis of the technical specifications by ensuring that adequate electrical power is available to operate the emergency safeguards equipment. By maximizing the operability of the LNP instrumentation without requiring high risk testing, the proposed change will improve the margin of safety as related to availability of electric power to safety related loads.

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

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

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

NRC Project Director: Phillip F. McKee

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

Date of amendment request: January 13, 1995

Description of amendment request: The proposed amendment would revise the Administrative Controls Section (6.0) of the Technical Specifications (TS) for Hope Creek Generating Station to reflect organizational changes and resultant management title changes. As indicated on the marked-up pages in Attachment 2, PSE&G requests that: 1) Vice President and Chief Nuclear Officer will be replaced with Chief Nuclear Officer and President - Nuclear Business Unit in TS 6.1.2, 6.2.1.c, 6.5.2.4.3.g, 6.5.2.4.4.a, 6.5.2.4.4.b, 6.5.2.6, 6.6.1.b, 6.7.1.a, and 6.7.1.c. 2) Vice President and Chief Nuclear Officer will be replaced with Vice President - Nuclear Operations in TS 6.5.1.8.b, and 6.5.1.9. 3) In addition, General Manager - Quality Assurance and Nuclear Safety will be replaced with Director - Quality Assurance and Nuclear Safety Review in TS 6.5.1.8.b, 6.5.1.9, 6.5.2.2, 6.5.2.4.3.g, 6.7.1.a, 6.7.1.c.

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

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

The proposed management title changes from Vice President and Chief Nuclear Officer to Chief Nuclear Officer and President - Nuclear Business Unit or Vice President - Nuclear Operations, and from General Manager - Quality Assurance and Nuclear Safety to Director - Quality Assurance and Nuclear Safety Review are administrative in nature and do not affect assumptions contained in the plant safety analysis, the physical design and/or operation of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The changes being proposed are purely administrative and will not lead to material procedure changes or to physical modifications. Therefore, the proposed changes do not create the possibility of a new or different type of accident from any accident previously evaluated.

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

The changes being proposed are administrative in nature and do not relate to or modify the safety margins defined in and maintained by the Technical Specifications. The changes discussed herein do not reduce the Technical Specification safety margin since all organizational responsibilities are being adequately implemented, and all personnel in place are properly qualified. 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: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

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

NRC Project Director: John F. Stolz

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

Date of amendment request: May 19, 1995 (TS 95-07)

Description of amendment request: The proposed change would (1) modify Surveillance Requirement (SR) 4.1.1.3 to allow suspension of the end of life (EOL) moderator temperature coefficient (MTC) surveillance measurement provided the benchmark criteria and the Revised Prediction as documented in the Core Operating Limits Report (COLR) are satisfied. The SR would also indicate that the data required for the calculation of the Revised Prediction is provided in the Most Negative Temperature Coefficient Limit Report per Specification 6.9.1.15. In addition, a grammatical error affecting the Unit 1 SR would be corrected; (2) modify Technical Specifications (TS) 6.9.1.14, COLR, by adding to the list of references: WCAP-13749-P-[A], ''Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement,'' May 1993 (Proprietary) (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient); (3) add Specification 6.9.1.15, which would require that the Most Negative MTC Report be prepared at least 60 days prior to the date the limit would become effective and be maintained on file. Also, the TS would require that the data required for the determination of the Revised Prediction of the 300 ppm/RTP MTC per WCAP-13749-P-[A] be included in the report.

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

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

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

The conditional exemption of the most negative moderator temperature coefficient (MTC) measurement does not change the most negative MTC surveillance requirement (SR) and limiting condition of operation (LCO) limits in the TSs. Since these MTC values are unchanged, and since the basis for the derivation of these values from the safety analysis moderator density coefficient (MDC) is unchanged, the constant MDC assumed for the Updated Final Safety Analysis Report (UFSAR) safety analyses will also remain unchanged. Therefore, no change in the modeling (i.e., probabilities) of the accident analysis conditions or response is necessary in order to implement the change to the conditional exemption methodology. In addition, since the constant MDC assumed in the safety analyses is not changed by the conditional exemption of the most negative MTC SR measurement, the consequences of an accident previously evaluated in the UFSAR are not increased. The dose predictions presented in the UFSAR for a steam generator tube rupture remain valid such that more severe consequences will not occur. Additionally, since mass and energy releases for a loss-of-coolant accident and a steamline break are not increased as a result of the unchanged MDC, the dose predictions for these events presented in the UFSAR also remain bounding.

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

Since the end-of-life MTC is not changed by the conditional exemption methodology of WCAP-13749-P, the possibility of an accident, which is different than any already evaluated in the UFSAR, has not been created. No new or different failure modes have been defined for any system or component nor has any new limiting single failure been identified. Conservative assumptions for the MDC have already been modeled in the UFSAR analyses. These assumptions will remain valid since the conditional exemption methodology documented in WCAP-13749-P does not change the safety analysis MDC nor the TS values of the MTC.

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

The conditional exemption methodology is documented in WCAP-13749-P. This WCAP has been evaluated (Reference: SECL 93-117,R1) relative to the design basis, including the TSs, and has been determined to bound the conditions under which the specifications permit operation. The results as presented in the UFSAR remain bounding since the MDC assumed in the safety analyses and the limiting conditions for operation and SR MTCs in the TSs remain unchanged. Therefore, the margin of safety, as defined in the bases to these TSs, is not reduced.

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

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

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

NRC Project Director: Frederick J. Hebdon

Tennessee Valley Authority, Docket No. 50-328, Sequoyah Nuclear Plant, Unit 2, Hamilton County, Tennessee

Date of amendment request: May 19, 1995 (TS 95-13)

Description of amendment request: The proposed change would revise License Condition 2.C.(17) to extend the required surveillance interval to May 4, 1996, for Surveillance Requirement 4.3.2.1.3. The proposed change would extend the Engineered Safety Features Response Time instrument tests required at 36-month intervals shown in Table 3.3-3 associated with safety injection, feedwater isolation, containment isolation Phase A, auxiliary feedwater pump, essential raw cooling water system, emergency gas treatment system, containment spray, containment isolation Phase B, turbine trip, 6.9-kilovolt shutdown board-degraded voltage or loss of voltage, and automatic switchover to containment sump actuations. The proposed extension will limit the interval past the allowable extension provided by TS 4.0.2 to 4.5 months.

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

TVA has evaluated the proposed technical specification (TS) change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c).

Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

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

The proposed change is temporary and allows a one-time extension of Surveillance Requirement 4.3.2.1.3 for Cycle 7 to allow surveillance testing to coincide with the seventh refueling outage. The proposed surveillance interval extension will not cause a significant reduction in system reliability nor affect the ability of the systems to perform their design function. Current monitoring of plant conditions and continuation of the surveillance testing required during normal plant operation will continue to be performed to ensure conformance with TS operability requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Extending the surveillance interval for the performance of specific testing will not create the possibility of a new or different kind of accidents. No changes are required to any system configurations, plant equipment, or analyses. Therefore, this change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

Surveillance interval extensions will not impact any plant safety analyses since the assumptions used will remain unchanged. The safety limits assumed in the accident analyses and the design function of the equipment required to mitigate the consequences of any postulated accidents will not be changed since only the surveillance test interval is being extended. Historical performance generally indicates a high degree of reliability, and surveillance testing performed during normal plant operation will continue to be performed to verify proper performance. Therefore, the plant will be maintained within the analyzed limits, and the proposed extension will not significantly reduce the margin of safety.

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

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

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

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: May 1, 1995

Brief description of amendments: The proposed amendment would: (1) reduce the minimum fuel oil volume requirement during MODES 5 and 6, for OPERABLE emergency diesel generators (EDG), and (2) allow continued OPERABLE status of diesel generators during all MODES, for 48 hours with greater than 6-day supply of diesel fuel for the associated diesel generator.

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. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

REDUCTION IN MINIMUM DIESEL FUEL STORED VOLUME WHILE SHUTDOWN

The first proposed change reduces the diesel fuel oil inventory required during plant shutdown conditions (MODES 5 and 6). The current fuel oil inventory requirement is the same for plant operation (MODES 1, 2, 3 and 4) and for plant shutdown. This current inventory requirement is based upon the seven days continuous operation of a diesel generator at its rated capacity which encompasses all load demands for the Loss of Coolant Accident concurrent with a Loss of Offsite Power (LOCA/LOOP) scenario. Because of reduced temperature and pressure, LOCA/LOOP is a less significant and probable event in MODES 5 and 6. The bounding scenario is considered to be a Loss of Offsite Power (LOOP) while the plant is shutdown (in MODES 5 and 6). The new diesel fuel oil inventory required during plant shutdown conditions is based on LOOP. Because this change only affects diesel fuel inventory, there is no impact on the probability of an accident. The consequences of LOOP event are unchanged since sufficient fuel remains available to allow the diesel generators to support mitigation of the event. Because seven days of fuel are required, there is no change in the consequences of any event which requires the diesel generators. Therefore, there is no significant increase in the probability or consequences of an accident previously evaluated as a result of this proposed change.

ADDITION OF REMEDIAL ACTION TO RESTORE THE STORED VOLUME OF DIESEL FUEL

The second proposed change applies to all MODES of operation. This change allows the diesel generator to remain OPERABLE if the fuel oil inventory falls below the minimum required in the storage system (i.e., fuel volume for 7-day operation of the diesel generator) but remains above a fuel volume for 6 days operation of the diesel generator. The minimum required fuel oil volume must be restored within 48 hours of falling below the limit. This relaxation by 48 hours allows sufficient time to replenish the required fuel oil volume and complete any required analysis prior to fuel oil addition to the storage tank. Because this change only affects diesel generator fuel inventory, there is no impact on the probability of an accident. Since the fuel oil replenishment can be obtained in less than six days after an event, there is no significant increase in the probability of a loss of all AC power (i.e., Station Blackout). Because the remaining fuel oil volume is larger than 6-day fuel supply and actions are initiated to obtain replenishment within this brief period, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

REDUCTION IN MINIMUM DIESEL FUEL STORED VOLUME WHILE SHUTDOWN

The first proposed change reduces the diesel fuel oil inventory required for plant shutdown conditions. As described above, LOOP is the limiting condition for diesel fuel oil inventory requirements for a plant in the shutdown condition. As the proposed fuel inventory is adequate for a shutdown LOOP and no hardware changes or system operation changes are involved, no new failure modes are introduced and hence, no new or different accidents from any previously evaluated are created.

ADDITION OF REMEDIAL ACTION TO RESTORE THE STORED VOLUME OF DIESEL FUEL

The second proposed change only affects diesel generator fuel inventory as well. There are no hardware changes and no changes in system operations involved; therefore, no new or different accidents from any accident previously evaluated are created.

3. Do the proposed changes involve a significant reduction in a margin of safety?

The intent of the Technical Specification is to conservatively assure sufficient fuel to assure diesel generator operation to support mitigation of postulated events. This intent is accomplished by conservatively assuring a seven day supply of fuel. Seven days fuel supply is considered sufficient to support the initial mitigation activities, identify the need for additional fuel, arrange for delivery, test and then add fuel to the storage tanks, if needed. The current diesel fuel oil inventory for operating conditions (MODES 1, 2, 3 and 4), is sufficient to conservatively support seven days of diesel generator operation for a LOCA with LOOP condition.

REDUCTION IN MINIMUM DIESEL FUEL STORED VOLUME WHILE SHUTDOWN

The proposed diesel fuel oil inventory for shutdown conditions (MODES 5 and 6), is adequate to conservatively support seven days of diesel generator operation for LOOP conditions. The proposed reduction in inventory between operating and shutdown conditions continues to support the different transient conditions which are applicable to the different modes of operation. Even though the minimum storage requirement during shutdown is being reduced, the basis of this specification continues to be conservatively satisfied and therefore this license amendment request does not involve a significant reduction in a margin of safety.

ADDITION OF REMEDIAL ACTION TO RESTORE THE STORED VOLUME OF DIESEL FUEL

The second proposed change which is applicable to all MODES of operation, allows 48 hours to restore diesel generator fuel oil inventory to the seven-day level as long as the inventory does not fall below the six-day level. The probability of a LOOP during this period is low. The 6-day fuel oil supply is calculated with adequate margin similar to the calculation of 7-day fuel oil inventory. In spite of the potential that there may be slightly less fuel available inlenishment within this brief period. Based on this and the low probability of an event during this brief period, it is considered that this change request does not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: George L. Edgar, Esq., Newman and Holtzinger, 1615 L Street, N.W., Suite 1000, Washington, D.C. 20036

NRC Project Director: William D. Beckner

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendments: May 4, 1994

Brief description of amendments: The amendments revise Limiting Condition for Operation (LCO) 3.4.8.3 and Surveillance Requirement 4.4.8.3.1, ''Overpressure Protection Systems.'' Specifically, the LCO and surveillance requirements are revised to clarify that both shutdown cooling system (SCS) suction relief valves shall be OPERABLE and aligned to provide overpressure protection not only during reactor coolant system (RCS) cooldown and heatup evolutions, but also during any steady-state temperature periods in the course of RCS cooldown or heatup evolutions.

Date of issuance: June 2, 1995

Effective date: June 2, 1995

Amendment Nos.: Unit 1 - Amendment No. 93; Unit 2 - Amendment No. 80; Unit 3 - Amendment No. 63

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

Date of initial notice in **Federal Register:** August 17, 1994 (59 FR 42333) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 2, 1995.No significant hazards consideration comments received: No.

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

Boston Edison Company, Docket No. 50-293, Pilgrim Nuclear Power Station,Plymouth County, Massachusetts

Date of application for amendment: November 22, 1994

Brief description of amendment: This amendment revises the suppression chamber water level operating range, increasing it 2 inches, and revises the water level recorder range in response to a commitment from an inspection.

Date of issuance: June 1, 1995

Effective date: June 1, 1995

Amendment No.: 163

Facility Operating License No. DPR-35: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 18, 1995 (60 FR 3672) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 1, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

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: February 24, 1995

Brief description of amendment: The proposed change would remove Section 4.3 from the Technical Specifications (TS) because the primary system testing following opening is already performed in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, as implemented in the licensee's inservice inspection program as required by TS 4.0.1.

Date of issuance: May 30, 1995Effective date: May 30, 1995

Amendment No.: 165

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

Date of initial notice in **Federal Register:** March 29, 1995 (60 FR 16183) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 30, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendments: October 15, 1992, as supplemented March 9, 1993.

Brief description of amendments: The amendments would modify the existing Dresden and Quad Cities Technical Specifications (TS) to format them in the style of the Boiling Water Reactor 4 (BWR) Standard Technical Specifications (STS). The amendments deal specifically with Section 3/4.4, ''Standby Liquid Control System (SLCS).''

Date of issuance: June 8, 1995

Effective date: For Dresden, immediately, to be implemented no later than December 31, 1995; for Quad Cities, immediately, to be implemented no later than June 30, 1996.

Amendment Nos.: 133, 127, 154, and 150

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

Date of initial notice in **Federal Register:** July 7, 1993 (58 FR 36429) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 8, 1995. No significant hazards consideration comments received: No

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

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

Date of application for amendments: April 10, 1995

Brief description of amendments: The amendments would change the Technical Specifications by (1) revising the low pressure value at which the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems can be tested to 150 psig, and (2) testing these systems against a system head corresponding to reactor vessel pressure when steam is supplied to the turbines at 920 psig to 1005 psig for high pressure testing and 150 psig to 325 psig for low pressure testing.

Date of issuance: May 30, 1995

Effective date: Immediately and shall be implemented within 60 days.

Amendment Nos.: 153 and 149

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

Date of initial notice in **Federal Register:** April 28, 1995 (60 FR 21009) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 30, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

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

Date of application for amendments: November 21, 1994.

Brief description of amendments: The amendments add footnotes in Limiting Condition for Operation 3.15.2.A of the Technical Specifications (TS) to allow a one-time extension of the allowed outage time (AOT) for an inoperable reserve offsite power source from 72 hours to 14 days. To provide additional assurance that redundant sources of power to the operating unit are operable during the AOT outage, the amendment also adds footnotes in Surveillance Requirement 4.15.2.A of the TS to modify the emergency diesel generator and the normal offsite power source testing requirements.

Date of issuance: May 31, 1995

Effective date: May 31, 1995

Amendment Nos.: 163 and 151

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

Date of initial notice in **Federal Register:** January 4, 1995 (60 FR 500). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 31, 1995.No significant hazards consideration comments received: No

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

Consumers Power Company, Docket No. 50-255, Palisades Plant, Van Buren County, MichiganDate of application for amendment: October 20, 1992

Brief description of amendment: This amendment revises Technical Specification 5.3.1a to account for changes being made to the Palisades Final Safety Analysis Report (FSAR) Section 4.2 following replacement of the steam generators.

Date of issuance: May 22, 1995

Effective date: May 22, 1995

Amendment No.: 166

Facility Operating License No. DPR-20. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 12, 1995 (60 FR 18624) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 22, 1995. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: January 13, 1995, as supplemented April 12 and 27, 1995

Brief description of amendment: This amendment revises the Technical Specifications to allow installed primary and secondary safety valve settings to be within a 3% tolerance of their nominal settings, but would require returning the valve settings to within 1% of the nominal settings if the valves are removed from the piping for maintenance or testing.

Date of issuance: June 8, 1995

Effective date: June 8, 1995

Amendment No.: 167

Facility Operating License No. DPR-20. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 1, 1995 (60 FR 11130) The April 12 and 27, 1995, letters provided clarifying information in response to the staff's request for additional information of April 11, 1995, and a telephone request for information on the Palisades loss of load analysis contained in the January 13, 1995, submittal. This information was within the scope of the original application and did not change the staff's initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 8, 1995.No significant hazards consideration comments received: No.

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

Detroit Edison Company, Docket No. 50-341, Fermi-2, Monroe County, Michigan Date of application for amendment: September 13, 1993

Brief description of amendment: The amendment revises Technical Specification (TS) 6.5.2.8 to relocate audit frequencies from the TS to the Quality Assurance Program located in Chapter 17.2 of the Updated Final Safety Analysis Report. A related change to extend the frequency of the use of an independent fire contractor to every third fire protection audit was denied.

Date of issuance: May 23, 1995

Effective date: May 23, 1995, with full implementation within 45 days.

Amendment No.: 104

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

Date of initial notice in **Federal Register:** April 12, 1995 (60 FR 18625) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 23, 1995. No significant hazards consideration comments received: No.

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

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

Date of amendment request: September 16, 1993

Brief description of amendment: The amendment revised the Technical Specifications by removing the incore detection system requirements. These requirements are to be relocated in the Updated Final Safety Analysis Report.

Date of issuance: May 30, 1995

Effective date: May 30, 1995, to be implemented within 60 days.

Amendment No.: 107

Facility Operating License No. NPF-38. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** October 27, 1993 (58 FR 57851) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 30, 1995. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: January 20, 1995

Brief description of amendments: These amendments will relocate the operability requirements for Incore Detectors in Technical Specification 3/4.3.3.2 to the Updated Final Safety Analysis Report, and revise Linear Heat Rate Surveillance 4.2.1.4, and Special Test Exceptions Surveillances 4.10.2.2, 4.10.4.2 (Unit 2 only), and 4.10.5.2, accordingly.

Date of issuance: June 6, 1995

Effective date: June 6, 1995

Amendment Nos.: 136 and 75

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

Date of initial notice in **Federal Register:** March 1, 1995 (60 FR 11132) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 6, 1995No significant hazards consideration comments received: No.

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

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

Date of application for amendments: December 29, 1994, as supplemented by letter dated May 2, 1995.

Brief description of amendments: The amendments revise TS 3/4.3, Instrumentation and its associated Bases, and TS 3/4.8, Electrical Power Systems to specify the appropriate actions to take in the event that an automatic load sequencer must be taken out of service or becomes inoperable.

Date of issuance: May 31, 1995

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

Amendment Nos.: 86 and 64

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

Date of initial notice in **Federal Register:** February 1, 1995 (60 FR 6301). The May 2, 1995, letter provided minor editorial changes that did not change the scope of the December 29, 1994, application and initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 31, 1995.No significant hazards consideration comments received: No

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

Gulf States Utilities Company, Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: January 13, 1993, as supplemented by letter dated October 18, 1993

Brief description of amendment: The amendment revised the River Bend Station, Unit 1 operating license to reflect a change in ownership of Gulf States Utilities (GSU). GSU, which ownes a 70 percent undivided interest in the River Bend Station, is a wholly-owned subsidiary company of Entergy Corporation. This amendment was originally issued on December 16, 1993, as License Amendment No. 69.

Date of issuance: June 8, 1995.

Effective date: June 8, 1995.

Amendment No.: 78

Facility Operating License No. NPF-47. The amendment revised the operating license.

Date of initial notice in **Federal Register:** July 7, 1993 (58 FR 36436) The October 18, 1993, supplemental letter provided clarifying information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 8, 1995.

Local Public Document Room location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803.

Gulf States Utilities Company, Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: January 13, 1993, as supplemented by letter dated June 29, 1993

Brief description of amendment: The amendment revised the River Bend Station, Unit 1 operating license to include as a licensee, Entergy Operations, Inc. (EOI), and to authorize EOI to use and operate River Bend and to possess and use related licensed nuclear materials. This amendment was originally issued on December 16, 1993 as License Amendment No. 70.

Date of issuance: June 8, 1995

Effective date: June 8, 1995

Amendment No.: 79

Facility Operating License No. NPF-47. The amendment revised the operating license.

Date of initial notice in **Federal Register:** July 7, 1993 (58 FR 36436) The June 29, 1993, supplemental letter provided clarifying information and did not change the initial no significant hazards consideration determination.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 8, 1995.No significant hazards consideration comments received. Yes. Comments and a request for hearing were received from Cajun Electric Power Cooperative of Baton Rouge, Louisiana.

Local Public Document Room location: Government Documents Department, Louisiana State University, Baton Rouge, Louisiana 70803.

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

Date of application for amendment: March 1, 1995

Brief description of amendment: The amendment revised the surveillance criteria for certain pumps and valves in the Low Pressure Coolant Injection (LPCI) subsystem; the Core Spray subsystems; and the Residual Heat Removal (RHR) Service Water, High Pressure Coolant Injection (HPCI), Emergency Service Water (ESW), and River Water Supply systems. The surveillance criteria changed from every three months to the testing frequency specified in the Inservice Testing program.

Date of issuance: May 18, 1995

Effective date: May 18, 1995

Amendment No.: 210

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

Date of initial notice in **Federal Register:** April 12, 1995 (60 FR 18626) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 18, 1995.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 10, 1995

Brief description of amendment: The amendment deletes Technical Specification Sections 3.7/4.7.H.3 to eliminate redundant Limiting Conditions of Operation and Surveillance Requirements for the containment hydrogen and oxygen analyzers.

Date of issuance: May 31, 1995

Effective date: May 31, 1995

Amendment No.: 211

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

Date of initial notice in **Federal Register:** April 26, 1995 (60 FR 20518) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 31, 1995.No significant hazards consideration comments received: No.

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

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

Date of amendment request: September 28, 1990

Brief description of amendment: The amendment revised the Technical Specifications to establish periodic operability testing of the reactor vessel overfill protection system. The changes were requested to satisfy a commitment in the licensee's response to Generic Letter 89-19, ''Request for Action Related to Resolution of Unresolved Safety Issue (USI) A-47.''

Date of issuance: June 8, 1995

Effective date: June 8, 1995

Amendment No.: 169

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

Date of initial notice in **Federal Register:** October 31, 1990 (55 FR 45885) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 8, 1995. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 31, 1995

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

Date of issuance: May 31, 1995

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

Amendment No.: 82

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

Date of initial notice in **Federal Register:** April 26, 1995 (60 FR 20520) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 31, 1995. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: January 23, 1995.

Brief description of amendment: The amendment revises the Technical Specifications to modify the containment spray system by replacing the present sodium hydroxide spray additive with the trisodium phosphate dodecahydrate pH control agent.

Date of issuance: May 26, 1995

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

Amendment No.: 115

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

Date of initial notice in **Federal Register:** March 1, 1995 (60 FR 11136). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 26, 1995.No significant hazards consideration comments received: No.

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

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

Date of application for amendments: September 20, 1994, as supplemented by letter dated April 14, 1995.

Brief description of amendments: The proposed amendments revise surveillance requirements (SRs) as recommended by NRC Generic Letter (GL) 93-05, ''Line-Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation'' of the combined Technical Specifications (TS) for the Diablo Canyon Nuclear Power Plant Unit Nos. 1 and 2. The specific TS changes are as follows:

(1) TS SR 4.1.3.1.2 is revised to change the frequency for testing the movability of the control rods from at least once per 31 days to at least once per 92 days.

(2) TS 3/4.3.2, Table 4.3-2, ''Engineered Safety Features Actuation System Instrumentation Surveillance Requirements,'' Functional Unit 3.c.4), and TS 3/4.3.3.1, Table 4.3-3, ''Radiation Monitoring Instrumentation for Plant Operations SRs,'' is revised to change the monthly channel functional test to quarterly.

(3) TS 3/4.5.1 is changed as follows: (a) TS SR 4.5.1.1a.1) is revised to more clearly state that the accumulator water volume and pressure must be verified to be within their limits. (b) TS SR 4.5.1.1b. is revised to specify that the boron concentration surveillance is not required to be performed if the accumulator makeup source was the refueling water storage tank (RWST). (c) TS SR 4.5.1.2 is relocated to plant procedures.

(4) TS SR 4.5.2c.2) is revised to clarify that a separate containment entry to verify the absence of loose debris is not required after each containment entry.

(5) TS SR 4.6.2.1d. is revised to change the frequency for a containment spray header flow test from at least once per 5 years to at least once per 10 years.

(6) TS SR 4.6.4.2a. is revised to change the verification of the minimum hydrogen recombiner sheath temperature from at least once per 6 months to at least once each refueling interval.

(7) TS SR 4.7.1.2.1 is revised to change the surveillance frequency for testing each auxiliary feedwater (AFW) pump from at least once per 31 days to at least once per 92 days on a staggered test basis.

(8) TS SR 4.10.1.2 is revised to lengthen the allowed period of time for a rod drop test from 24 hours to 7 days prior to reducing shutdown margin to less than the limits of TS 3.1.1.1.

(9) TS SR 4.11.2.6 is revised to change the surveillance frequency from 24 hours to 7 days when radioactive material is being added to the gas decay tanks and to add a requirement to monitor radioactive material concentrations in the gas decay tanks at least once per 24 hours when system degassing operations are in progress.

Date of issuance: May 26, 1995

Effective date: May 26, 1995, to be implemented within 60 days of issuance.

Amendment Nos.: Unit 1 - Amendment No. 102; Unit 2 Amendment No. 101

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

Date of initial notice in **Federal Register:** October 26, 1994 (59 FR 53843) The April 14, 1995, letter provided clarifying information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 26, 1995.No significant hazards consideration comments received: No.

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

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

Date of application for amendments: December 30, 1994 (LAR 94-12)

Brief description of amendments: These amendments clarify the technical specifications (TS) issued in license amendments 84/83 associated with the Eagle 21 reactor protection system modification, delete TS references to RM-14A and RM-14B, remove cycle-specific TS requirements, and incorporate editorial corrections.

Date of issuance: June 2, 1995

Effective date: June 2, 1995, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 1 - Amendment No. 103; Unit 2 - Amendment No. 102

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

Date of initial notice in **Federal Register:** March 15, 1995 (60 FR 14026) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 2, 1995.No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 6, 1995, as supplemented by letters dated March 23, 1995, and May 22, 1995.

Brief description of amendments: The amendments would allow the storage of fuel with enrichments up to and including 5.0 weight percent U-235, would clarify that substitution of fuel rods with filler rods is acceptable for fuel designs that have been analyzed with applicable NRC-approved codes and methods, and would allow the use of ZIRLO fuel cladding in the future in addition to Zircaloy-4.

Date of issuance: June 7, 1995

Effective date: June 7, 1995, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 1 - Amendment No. 104; Unit 2 - Amendment No. 103

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

Date of initial notice in **Federal Register:** March 1, 1995 (60 FR 11138) The licensee's supplemental letters provided additional clarifying information. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 7, 1995. No significant hazards consideration comments received: Yes. Comments were submitted by Jill ZamEk on behalf of the San Luis Obispo Mothers for Peace by letter dated March 30, 1995.

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

Pacific Gas and Electric Company, Docket No. 50-133, Humboldt Bay Power Plant, Unit 3, Humboldt County, California

Date of application for amendment: November 23, 1994, as supplemented April 27, 1995.

Brief description of amendment: This amendment revised the Technical Specifications Section VII.C., Plant Staff, to decrease the minimum staff requirements for the shift operating organization from five to two persons.

Date of issuance: May 31, 1995

Effective date: This license amendment is effective as of the date of its issuance and must be fully implemented no later than 30 days from the date of issuance.

Amendment No.: 28Facility License No. DPR-7: The amendment revised the TS.

Date of initial notice in **Federal Register:** March 1, 1995 (60 FR 11139) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 31, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Humboldt County Library, 636 F Street, Eureka, California 95501.

PECO Energy Company, Public Service Electric and Gas CompanyDelmarva Power and Light Company, and Atlantic City Electric Company,Docket Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: February 10, 1995

Brief description of amendments: These amendments correct administrative errors in Section 4.11.A of the Technical Specifications (TSs). The errors were made in the TSs by Amendments 9 and 7 dated June 25, 1975.

Date of issuance: May 30, 1995

Effective date: May 30, 1995

Amendments Nos.: 202 and 205

Facility Operating License Nos. DPR-44 and DPR-56: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 26, 1995 (60 FR 20521) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 30, 1995.No significant hazards consideration comments received: No

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

PECO Energy Company, Public Service Electric and Gas CompanyDelmarva Power and Light Company, and Atlantic City Electric Company,Docket Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station,Unit Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: September 26, 1994

Brief description of amendments: These amendments extend the surveillance test intervals and allowable out-of service times for the testing and or repair of instrumentation that actuate the Reactor Protection System, Primary Containment Isolation, Core and Containment Cooling systems, Control Rod Blocks, Radiation Monitoring systems and Alternate Rod Insertion/ Recirculation Pump Trip.

Date of issuance: June 6, 1995

Effective date: June 6, 1995

Amendments Nos.: 203 and 206

Facility Operating License Nos. DPR-44 and DPR-56: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 15, 1995 (60 FR 14027) The supplemental letters dated January 5, and March 23, 1995, provided clarifying information and did not change the initial proposed no significant hazards consideration determination.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 6, 1995.No significant hazards consideration comments received: No

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

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: April 30, 1993

Brief description of amendments: These amendments changed the Technical Specifications by deleting Section 3/4.3.8 of the Turbine Overspeed Protection System.

Date of issuance: June 1, 1995

Effective date: June 1, 1995

Amendment Nos.: 146 and 116

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

Date of initial notice in **Federal Register:** June 9, 1993 (58 FR 32389) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 1, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendments: August 25, 1993, as supplemented by letters dated June 27, 1994, and May 5, 1995

Brief description of amendments: These amendments modify Technical Specification Surveillance Requirement 4.7.1.3 to require that all spray pond spray network piping above the frost line be drained at an ambient temperature below 40°F, and within 1 hour after being used only when the ambient air temperature is below 40°F.

Date of issuance: June 1, 1995

Effective date: June 1, 1995

Amendment Nos.: 90 and 54

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

Date of initial notice in **Federal Register:** September 29, 1993 (58 FR 50972) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 1, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendment: August 30, 1994

Brief description of amendment: The changes relocate Technical Specification (TS) 3.3.7.9, Loose Parts Detection System (LPDS), Surveillance Requirement 4.3.7.9, and associated Bases from the TSs to the Updated Final Safety Analysis Report. The TS index is also revised by removing the reference to LPDS.

Date of issuance: May 25, 1995

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

Amendment No.: 73

Facility Operating License No. NPF-57: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 29, 1995 (60 FR 16197) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 25, 1995. No significant hazards consideration comments received: No

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

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: February 9, 1995

Brief description of amendments: The amendments revise the Administrative

Controls section of the Technical Specifications to reflect organizational changes and resultant management title changes.

Date of issuance: June 6, 1995

Effective date: June 6, 1995

Amendment Nos.: 168 and 150

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

Date of initial notice in **Federal Register:** March 29, 1995 (60 FR 16200) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 6, 1995.No significant hazards consideration comments received: No

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

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 application for amendments: March 6, 1995

Brief description of amendments: The amendments relocate the seismic and meteorological monitoring instrumentation from the Technical Specifications to the Final Safety Analysis Report in accordance with the ''Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors,'' dated July 22, 1993.

Date of issuance: May 22, 1995

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

Amendment Nos.: 115 and 107

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

Date of initial notice in **Federal Register:** April 12, 1995 (60 FR 18628) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 22, 1995.No significant hazards consideration comments received: No

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

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

Date of application for amendment: April 6, 1995 (TS 95-09)

Brief description of amendment: The amendment modifies Operating License Condition 2.C.(25) to provide a limited extension of the ice condenser surveillance test interval on Unit 1 to coincide with the Cycle 7 refueling outage.

Date of issuance: May 30, 1995

Effective date: May 30, 1995

Amendment No.: 200

Facility Operating License Nos. DPR-77: Amendment revises the technical specifications.

Date of initial notice in **Federal Register:** April 26, 1995 (60 FR 20526) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 30, 1995.No significant hazards consideration comments received: None

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

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

Date of application for amendments: April 6, 1995

Brief description of amendments: The amendments revise the surveillance requirement for the power range neutron flux channel calibration frequency from monthly to every 31 effective full power days and delays first performance of the surveillance after reaching 15 percent power for 96 hrs.

Date of issuance: May 30, 1995

Effective date: May 30, 1995

Amendment Nos.: 199 and 190

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

Date of initial notice in **Federal Register:** April 26, 1995 (60 FR 20530) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 30, 1995.No significant hazards consideration comments received: None

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

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

Date of application for amendments: April 6, 1995

Brief description of amendments: The amendments revise the definition of core alteration, quadrant power tilt ratio, and modifies the operational mode parameters table in the Unit 1 technical specifications.

Date of issuance: June 1, 1990

Effective date: June 1, 1990

Amendment Nos.: 201 and 191

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

Date of initial notice in **Federal Register:** April 26, 1995 (60 FR 20531) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 1, 1995.No significant hazards consideration comments received: None

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

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

Date of amendment request: February 14, 1994 (TXX-94046 LAR 94-006)

Brief description of amendments: The proposed changes would revise the Technical Specifications (TSs) for Comanche Peak Steam Electric Station, Units 1 and 2 in the following three areas: 1) a change to the allowable value for the Unit 2 pressurizer pressure-low and Unit 2 overtemperature N-16 (OTN-16) reactor trip setpoints; 2) an administrative change to delete an option which allowed continued operation for a period of time when a reactor trip system (RTS) or engineered safety features actuation system (ESFAS) instrumentation or interlocks trip setpoint is found less conservative than the allowable value; and 3) an administrative change to combine the Unit 1 and Unit 2 line items for RTS or ESFAS trip setpoint and allowable values which are the same.

Date of issuance: May 31, 1995

Effective date: May 31, 1995, to be implemented within 30 days.

Amendment Nos.: Unit 1 - Amendment No. 41; Unit 2 - Amendment No. 27

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

Date of initial notice in **Federal Register:** June 22, 1994 (59 FR 32238) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 31, 1995.No significant hazards consideration comments received: No.

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

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

Date of amendment request: March 24, 1995

Brief description of amendment: The amendment relaxes the requirement to sample the accumulator after refilling from the RWST.

Date of issuance: May 30, 1995

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

Amendment No.: Amendment No. 87

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

Date of initial notice in **Federal Register:** April 12, 1995 (60 FR 18632) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 30, 1995.No significant hazards consideration comments received: No. Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621.

Dated at Rockville, Maryland, this 14th day of June, 1995.

For the Nuclear Regulatory Commission

John N. Hannon,

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

[Doc. 95–15057 Filed 6–20–95; 8:45]

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[Docket No. 50–255]

Consumers Power Company (Palisades Plant); Exemption

I

Consumers Power Company (CPCo, the licensee) is the holder of Facility Operating License No. DPR–20 which authorizes operation of the Palisades Plant, a pressurized water reactor (PWR) located in Van Buren County, Michigan. The license provides, among other things, that the facility is subject to all rules, regulations, and orders of the Nuclear Regulatory Commission (the Commission) now or hereafter in effect.

II

Pursuant to 10 CFR 50.12(a), the NRC may grant exemptions from the requirements of the regulations (1) which are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security; and (2) where special circumstances are present.

Section III.D.1.(a) of Appendix J to 10 CFR part 50 requires the performance of three Type A containment integrated leakage rate tests (ILRTs), at approximately equal intervals during each 10-year service period of the primary containment. The third test of each set shall be conducted when the plant is shut down for the 10-year inservice inspection of the primary containment.

III

By letter dated March 17, 1995, as supplemented April 26, 1995, CPCo requested temporary relief from the requirement to perform a set of three Type A tests at approximately equal intervals during each 10-year service period of the primary containment. The requested exemption would permit a one-time interval extension of the third Type A test by approximately 21 months (from the 1995 refueling outage, currently scheduled to begin in May 1995, to the 1997 refueling outage) and would permit the third Type A test of the second 10-year inservice inspection period to not correspond with the end of the current American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) inservice inspection interval.

The licensee's request cites the special circumstances of 10 CFR 50.12, paragraphs (a)(2) (ii) and (iii), as the basis for the exemption, and states that the exemption would eliminate a cost of $1 million for the Type A test which is not necessary to achieve the underlying purpose of the rule. 10 CFR part 50 Appendix J, states that the purpose of the Type A, B, and C tests is to assure that leakage through the primary containment shall not exceed the allowable leakage rate values as specified in the technical specifications or associated bases. CPCo points out that the existing Type B and C testing programs are not being modified by this request and will continue to effectively detect containment leakage caused by the degradation of active containment isolation components as well as containment penetrations. It has been the experience at the Palisades Plant that, with the exception of the 1978 test results, during the six Type A tests conducted from 1974 to date, any significant containment leakage paths are detected by the Type B and C testing. The Type A test results have only been confirmatory of the results of the Type B and C test results. The testing history, structural capability of the containment, and the risk assessment establish that there is significant assurance that the extended interval between Type A tests will not adversely impact the leak-tight integrity of the containment and that performance of the Type A test is not necessary to meet the underlying purpose of Appendix J. The licensee also references the proposed revision to Appendix J which would reduce the frequency of Type A tests.

IV

Section III.D.1.(a) of Appendix J to 10 CFR part 50 states that a set of three Type A leakage rate tests shall be performed at approximately equal intervals during each 10-year service period.

The licensee proposes an exemption to this section which would provide a one-time interval extension for the Type A test by approximately 21 months. The Commission has determined, for the reasons discussed below, that pursuant to 10 CFR 50.12(a)(1) this 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. The Commission further determines that special circumstances, as provided in 10 CFR 50.12(a)(2) (ii) and (iii), are present justifying the exemption; namely, that application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule and would impose excessive cost.

The underlying purpose of the requirement to perform Type A containment leak rate tests at intervals during the 10-year service period is to ensure that any potential leakage pathways through the containment boundary are identified within a time span that prevents significant degradation from continuing or becoming unknown. The NRC staff has reviewed the basis and supporting information provided by the licensee in the exemption request. The NRC staff has noted that the licensee has a good record of ensuring a leak-tight containment following the submittal of its Corrective Action Plan on June 30, 1986. The Corrective Action Plan was submitting following three consecutive Type A test failures, of which one was the 1978 test failure. However, the licensee has noted that the containment penetration local leak rate tests (LLRT, Type B and C tests) accounted for the majority of the before maintenance adjustment to the as-found ILRT (Type A) results during the as-found test failures. The penetration associated with the 1978 test failure was significantly modified in the mid-1980's to improve the LLRT test configuration to properly monitor the entire penetration boundary. In addition, the licensee aggressively replaced or repaired the valves and penetrations that accounted for the as-found test failures, with no repeat occurrences.

The NRC staff reviewed the LLRT Corrective Action Plan and granted an