Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

A portion of this meeting may be closed to discuss the proposed NRC thermal hydraulic research program budget. Discussion of the impact of possible budget reduction on continuing and proposed research contracts, if held in public session, might result in premature disclosure of information which would in turn frustrate the Commission's ability to effectively implement the affected programs.

The agenda for the subject meeting shall be as follows:

Wednesday, September 18, 1996—8:30 a.m. until the conclusion of business
Thursday, September 19, 1996—8:30 a.m. until the conclusion of business

The Subcommittee will: (1) Begin its review of the NRC–RES Program to revise/replace the current suite of NRC–RES thermal hydraulic codes and (2) discuss the status of the RES thermal hydraulic research program and associated budget. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, its consultants, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the scheduling of sessions which are open to the public, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting the cognizant ACRS staff engineer, Mr. Paul A. Boehnert (telephone 301/415–8065) between 7:30 a.m. and 4:15 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Dated: August 21, 1996.

Sam Duraiswamy,

Chief, Nuclear Reactors Branch.

[FR Doc. 96–21940 Filed 8–27–96; 8:45 am]

BILLING CODE 7590–01–P

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from August 3, 1996, through August 16, 1996. The last biweekly notice was published on August 14, 1996 (61 FR 42274).

Notice Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

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

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

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

Written comments may be submitted by mail to the Chief, Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and should cite the publication date and page number of this Federal Register notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By September 27, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for

Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (Project Director): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the attorney for the licensee.

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

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

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

Date of amendment request: July 19, 1996

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 3/4.6.2, Containment Spray System, to extend the surveillance interval for performance of an air or smoke flow test through containment spray nozzles from once per 5 years to once per 10 years. This change is consistent with the guidance in NRC Generic Letter 93-05, ''Line Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operations,'' and NUREG-1366, ''Improvements To Technical Specifications Surveillance Requirements.''

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

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

The proposed extended testing frequency of containment spray nozzles will not affect any initiators of any previously evaluated accidents or change the manner of operation for any system or component. The containment spray system serves a mitigating function by removing heat and fission products from a post accident containment atmosphere. Increasing the surveillance test interval will not affect the system's ability to provide this function. Therefore, there would be no increase in the probability or consequences of an accident previously evaluated.

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

Since the proposed change affects only a surveillance frequency, it will not involve any physical alterations to plant equipment or alter the manner in which any safety-related system performs its function. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not affect any Final Safety Analysis Report (FSAR) Chapter 15 accident analyses or impact the margin of safety for the containment spray system as defined in the Bases to the Technical Specifications. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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

Attorney for licensee: William D. Johnson, Vice President and Senior Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: Eugene V. Imbro

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

Date of amendment request: June 10, 1996

Description of amendment request: To change the technical specifications to reflect the transition from General Electric Company (GE) to Siemens Power Corporation (SPC) as the fuel supplier for the Quad Cities Nuclear Power Station, Units 1 and 2.

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

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

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits will be established consistent with NRC approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. The proposed Technical Specifications amendment reflects previously approved SPC methodology used to analyze normal operations, including anticipated operational occurrences (AOOs), and to determine the potential consequences of accidents.

Licensing Methods and Models

The proposed amendment is to support operation with NRC approved fuel and licensing methods supplied from Siemens Power Corporation. In accordance with FSAR Chapter 15, the same accidents and transients will be analyzed with the new fuel and methods as were analyzed by GE for GE fuel. The analysis methods and models are NRC approved. These approved methods and models are used to determine the fuel thermal limits (e.g., LHGR, APLHGR, MCPR). The SPC core monitoring code enables the site to monitor k_{eff} as well as rod density to perform the reactivity anomaly surveillance. This is consistent with GE methodology. The support systems for minimizing the consequences of transients and accidents are not affected by the proposed amendment. Therefore, the change in licensing analysis methods and models does not significantly increase the probability of an accident or the consequences of an accident previously identified.

New Fuel Design

The use of ATRIUM 9B fuel at Quad Cities does not involve a significant increase in the probability or consequences of any accident previously evaluated in the FSAR. The ATRIUM-9B fuel is generically approved for use as a reload BWR fuel type (Reference: ANF-89-014(P)(A) Rev. 1 Supplement 1, General Mechanical Design for Advanced Nuclear Fuels 9X9-IX and 9X9-9X BWR Reload Fuel). Limiting postulated occurrences and normal operation have been analyzed using NRC-approved methods for the ATRIUM 9B fuel design to ensure that safety limits are protected and that acceptable transient and accident performance is maintained.

The reload fuel has no adverse impact on the performance of in-core neutron flux instrumentation or CRD response. The ATRIUM-9B fuel design will not adversely affect performance of neutron instrumentation nor will it adversely affect the movement of control blades relative to the GE fuel. The exterior dimensions of the ATRIUM-9B fuel have been evaluated by ComEd; the SPC fuel provides adequate clearances relative to the GE10 fuel installed at Quad Cities. Thus, no increased interactions with the adjacent control blade and nuclear instrumentation are created. Additionally, given the above mentioned overall envelope similarities, no problems are anticipated with other station equipment such as the fuel storage racks, the new fuel inspection stand and the spent fuel pool fuel preparation machine. Therefore, the probability of adverse interactions between the Siemens fuel and components in the core and fuel handling equipment is not significantly increased.

The ATRIUM 9B design is neutronically compatible with the existing fuel types and core components in the Quad Cities core. SPC tests have demonstrated that the ATRIUM-9B fuel design is hydraulically compatible with the GE9/GE10 fuel. The bundle pressure drop characteristics of the ATRIUM 9B bundle are similar to those of the GE9/GE10 fuel design, hence core thermal-hydraulic stability characteristics are not adversely affected by the ATRIUM 9B design. Cycle stability calculations are performed by SPC. Therefore, the probability of thermal hydraulic instability is not significantly increased.

An evaluation of the Emergency Procedures is being performed to ensure that the use of the ATRIUM-9B fuel at Quad Cities does not alter any assumptions previously made in evaluating the radiological consequences of an accident at Quad Cities Station. Therefore, the radiological consequences of accidents are not significantly increased.

Methods approved by the NRC are being used in the evaluation of fuel performance during normal and abnormal operating conditions. The ComEd and SPC methods to be used for the cycle specific transient analyses have been previously NRC approved. The proposed methodologies are administrative in nature and do not significantly affect any accident precursors or accident results; as such, the proposed incorporation of the SPC methodologies for Quad Cities does not significantly increase the probability or consequences of any previously evaluated accidents. The description of the fuel is modified to include the water box design of the NRC approved ATRIUM-9B fuel. This change is administrative.

Review of the above concludes that the probability of occurrence and the consequences of an accident previously evaluated in the safety analysis report have not been significantly increased.

* * * * *

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

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in allowable modes of operation.

Licensing Methods and Models

The proposed Technical Specification amendment reflects previously approved SPC methodology used to analyze normal operations, including AOOs, and to determine the potential consequences of accidents. In accordance with FSAR Chapter 15, the same accidents and transients will be analyzed with the new fuel and methods as were analyzed by GE for GE fuel. As stated above, the proposed changes do not permit modes of operation which differ from those currently permitted; therefore, the possibility of a new or different kind of accident is not created. Plant support equipment is not affected by the proposed changes; therefore, no new failure modes are created.

New Fuel Design

The basic design concept of a 9x9 fuel pin array with an internal water box has been used in various lead assembly programs and in reload quantities in Europe since 1986.

WNP-2 has loaded reload quantities since 1991. Approximately 650 water box assemblies have been irradiated in the United States through 1995, with a substantially higher number being irradiated overseas. The NRC has reviewed and approved the ATRIUM-9B fuel design (Reference: ANF-89-014(P)(A) Rev. 1 Supplement 1, Generic Mechanical Design for Advanced Nuclear Fuels 9X9-IX and 9X9-9X BWR Reload Fuel). The similarities in fuel design and operation between GE and SPC, and the previous Boiling Water Reactor experience with both vendors' fuel indicate there would be no new or different types of accidents for Quad Cities than have been considered for the existing fuel. Therefore, the use of ATRIUM-9B fuel at Quad Cities does not create the possibility of a new or different kind of accident from any accident previously evaluated.

* * * * *

3) Involve a significant reduction in the margin of safety for the following reasons:

The existing margin to safety is provided by the existing acceptance criteria (e.g., 10CFR50.46 limits). The proposed Technical Specification amendment reflects previously approved SPC methodology used to demonstrate that the existing acceptance criteria are satisfied. The revised methodology has been previously reviewed and approved by the USNRC for application to reload cores of GE BWRs. References for the Licensing Topical Reports which document this methodology, and include the Safety Evaluation Reports prepared by the USNRC, are added to the Reference section of the Technical Specifications as part of this amendment.

Licensing Methods and Models

The proposed amendment does not involve changes to the existing operability criteria. NRC approved methods and established limits (implemented in the COLR) ensure acceptable margin is maintained. The ComEd and SPC reload methodologies for the ATRIUM-9B reload design are consistent with the Technical Specification Bases. The Limiting Conditions for Operation are taken into consideration while performing the cycle specific and generic reload safety analyses. NRC approved methods are listed in Section 6 of the Technical Specifications.

Analyses performed with NRC-approved methodology have demonstrated that fuel design and licensing criteria will be met during normal and abnormal operating conditions. The same margins of safety are utilized by SPC as GE (e.g., limits on peak cladding temperature, cladding oxidation, plastic strain). Therefore, there is not a significant reduction in the margin of safety.

New Fuel Design

The exterior dimensions of the ATRIUM-9B fuel assembly result in equivalent clearances relative to the GE10B. Thus, no increased interactions with the adjacent control blade and nuclear instrumentation are created. The change does not adversely impact equipment important to safety; 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 requested amendments involve no significant hazards consideration.

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

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

NRC Project Director: Robert A. Capra

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

Date of amendment request: August 12, 1996

Description of amendment request: The proposed change would implement the performance-based containment leak rate testing provisions of Option B to 10 CFR Part 50 Appendix J for the Type A (containment) testing program.

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

The following analysis is presented, pursuant to 10 CFR 50.91, to demonstrate that the proposed change will not create a Significant Hazard Consideration.

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 change does not affect reactor operations or accident analysis, and has no significant radiological consequences. Therefore, this proposed change will not involve an 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, or change any design basis. The proposed changes will not affect the response of [the] containment during a design basis accident.

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

The proposed changes are based on NRC-accepted provisions, and maintain necessary levels of reliability of containment integrity. The performance-based approach to leakage rate testing recognizes that historically good results of containment testing provide appropriate assurance of future containment integrity; this supports the conclusion that the impact on the health and safety of the public as a result of extended test intervals is negligible.

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

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: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: May 31, 1996

Description of amendment request: The proposed amendment revises the surveillance test interval for the reactor protection system reactor trip breakers, reactor trip modules, and electronic trip relays from 1 month to 6 months. In addition to requesting a change to the Arkansas Nuclear One, Unit 1 Technical Specifications, the request also proposes the same changes to NUREG-1430, Standard Technical Specifications - Babcock and Wilcox Plants.

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

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

The accident mitigation features of the plant are not affected by the proposed test interval extension. The results of the B&W Owners Group Topical Report BAW-10167, Supplement 3, ''Justification for increasing The Reactor Trip System On-Line Test Intervals,'' show that the test interval extension of the reactor protection system trip devices is not a significant contributor to trip system unavailability or the risk of core damage.

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

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

The reactor trip device surveillance test interval is not, in and of itself, considered to be an accident initiator. Failure of a trip device to function is an analyzed condition and does not constitute a new or different kind of accident.

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

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

The results of the B&W Owners Group Topical Report BAW-10167, Supplement 3, ''Justification for Increasing The Reactor Trip

System On-Line Test Intervals,'' show that the test interval extension of the reactor protection system trip devices is not a significant contributor to trip system unavailability or the risk of core damage. In addition, the uncertainty analysis contained in BAW-10167 confirms the robustness of the results by demonstrating that even with an order of magnitude change in the failure data, the incremental increase due to an increased test interval is insignificant. Entergy Operations has reviewed BAW-10167 and found it applicable to ANO-1.

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

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

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

NRC Project Director: William D. Beckner

Entergy Operations, Inc., Docket Nos. 50-313 and 50-368, Arkansas Nuclear One, Unit Nos. 1 and 2 (ANO-1&2), Pope County, Arkansas

Date of amendment request: May 9, 1996

Description of amendment request: The proposed amendment changes the name of Arkansas Power and Light Company (AP&L) to Entergy Arkansas, Inc. in both the Operating License and the Technical Specifications. AP&L is licensed to own and possess Arkansas Nuclear One (ANO). The company licensed to operate ANO, Entergy Operations, Inc. is unaffected by this change.

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 Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The proposed change documents changing the legal name of the company. The proposed change will not affect any other obligations. The company will continue to own all of the same assets, will continue to serve the same customers, and will continue to honor all existing obligations and commitments. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The administrative changes in the operating license requirements do not involve any change in the design of the plant. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change is administrative in nature and does not reduce the margin of safety imposed by any current requirements. Therefore, this change does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: William D. Beckner

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

Dates of amendment request: July 17, 1996

Description of amendment request: The licensee proposed to change the Turkey Point Units 3 and 4 Technical Specifications (TS) to implement 10 CFR 50, Appendix J, Option B, for containment leakage testing. Changes include relocating the details for containment testing to the ''containment leakage rate testing program'' and adding the requirements of the containment leakage rate testing program to TS 6.8.4, which describes facility programs.

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.

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

a) These proposed changes are all consistent with NRC requirements and guidance for implementation of 10 CFR 50, Appendix J, Option B.

b) Based on industry and NRC evaluations performed in support of developing Option B, these changes potentially result in a minor increase in the consequences of an accident previously evaluated due to the expanded testing intervals. However, the proposed changes do not result in an increase in the core damage frequency since the containment system is used for mitigation purposes only.

c) These changes are expected to result in increased attention to components with poor leakage test history as part of the performance-based nature of Option B, such that the marginally increased consequences from the expanded testing intervals may be further reduced or negated.

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

The use of the modified specifications can not create the possibility of a new or different kind of accident from any previously evaluated since the proposed amendments will not change the physical plant or the modes of plant operation defined in the facility operating license. No new failure mode is introduced due to the implementation of a performance-based program for containment leakage rate testing, since the proposed changes do not involve the addition or modification of equipment, nor do they alter the design or operation of affected plant systems, structures, or components.

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

The operating limits and functional capabilities of the affected systems, structures, and components are basically unchanged by the proposed amendments due to the following reasons:

a) The acceptance criteria for total integrated containment leakage of 1.0 L_a is consistent with the current technical specifications and is within the design basis accident assumptions, and therefore does not reduce the margin of safety.

b) The increase in intervals between leak-test surveillances will not significantly reduce the margin of safety as shown by findings in NUREG 1493, ''Performance-Based Containment Leak-Test Program'', which was based on implementation of the performance-based testing of Option B.

Therefore these changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Frederick J. Hebdon

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: May 21, 1996

Description of amendment request: The proposed change to the condensate storage tank (CST) level indication would ensure that the water level is sufficient to provide 50,000 gallons of water for core spray makeup to the reactor pressure vessel.

Technical Specification (TS) Surveillance Requirement (SR) 3.5.2.2.b for ECCS - Shutdown states: ''Condensate storage tank (CST) water level is [greater than or equal to] 12 feet.'' The corresponding Bases state: ''... the CST contains [greater than or equal to] 150,000 gallons of water, equivalent to 12 feet, ensures that the CS System can supply at least 50,000 gallons of makeup water to the RPV.''

Subsequent licensee analyses confirmed that Plant Hatch Units 1 and 2 CST configurations are different; that is, for both CSTs, a water level of 12 feet is not equivalent to the required capacity of 150,000 gallons of water. Based on these calculations, the correct level for the Unit 1 CST is 13 feet, and the correct level for the Unit 2 CST is 15 feet.

The proposed change would revise Unit 1 and Unit 2 SR 3.5.2.2.b to require a CST water level of greater than or equal to 13 feet and greater than or equal to 15 feet, respectively, to ensure at least 50,000 gallons of water are available for core spray (CS) makeup to the reactor pressure vessel (RPV).

The associated Bases for each unit will be revised accordingly.

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

1. The proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated, because this administrative change to the CST water level does not alter the operation of any plant system or component. The proposed change does not involve a physical modification to any structure, system, or component. The minimum CST water level for each unit is being increased to account for the height of the CS suction standpipe within each CST and the differences in the Unit 1 and

Unit 2 CST diameters (gallons/ft of water) as follows:

a. Unit 1 - The proposed minimum water level is calculated as: CS suction standpipe height of 9 ft + (50,000 gallons divided by 12,704 gallons/ft) = 12.93 ft or 13 ft.

b. Unit 2 - The proposed minimum water level is calculated as: CS suction standpipe height of 10 ft + (50,000 gallons divided by 11,343 gallons/ft) = 14.4 ft or 15 ft.

The revised minimum levels ensure at least 50,000 gallons of water are provided above the top of the standpipe in each unit's CST and are available for CS makeup to the RPV, as stated in the applicable Bases. The TS Limiting Conditions for Operation (LCO) remain unaffected by the proposed change.

2. The proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated. Revising Surveillance Requirement acceptance criteria does not result in any physical modification to the plant or operation of any existing equipment.

3. The proposed TS change does not involve a significant reduction in a margin of safety, since this administrative change only ensures the existing TS Bases are satisfied by increasing the minimum CST water level requirement to ensure at least 50,000 gallons of water are available for CS injection to the RPV. The proposed change does not involve a physical modification to any structure, system or component, and does not modify the operation of any existing equipment.

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

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: July 8, 1996

Description of amendment request: The proposed amendment would clarify that the component cooling water system surge tank level instrumentation can be demonstrated operable by performing a channel calibration test during any plant mode of operation.

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

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

The proposed change to Technical Specification Surveillance Requirement 4.7.3.b.3 will not effect any accident initiators or precursors and will not alter the design assumptions for the systems or components used to mitigate the consequences of an accident. Calibration is performed on level instrumentation of Component Cooling Water System trains that are out of service for scheduled maintenance. Isolation redundancy is provided by instrumentation associated with the trains that are in service during the calibration. Since the surveillance will continue to be performed at the specified interval, this proposed change will not increase the probability of occurrence of an accident previously evaluated. The surveillance does not differ from those previously performed; therefore, there is no impact on the consequences of an accident previously evaluated.

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

Clarifying the surveillance interval for surge tank level instrumentation does not involve installation or operation of new or different kinds of equipment. There is no change in the procedures as described in the Technical Specifications. The change only clarifies the interval at which the subject calibration will be performed. 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 specified surveillance will remain as stated in the Technical Specifications. Consequently, there is no reduction in the effectiveness of the surveillance in ensuring equipment operability. Calibration is performed on level instrumentation of Component Cooling Water System trains that are out of service for scheduled maintenance. Isolation redundancy is provided by instrumentation associated with the trains that are in service during the calibration. Consequently, clarifying the interval at which the calibration is performed will have no significant impact on the margin of safety.

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

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

Attorney for licensee: Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, N.W., Washington, DC 20036-5869

NRC Project Director: William D. Beckner

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: August 8, 1996

Description of amendment request: The proposed amendment would allow the transition from Mode 4 to Mode 3 with the turbine-driven auxiliary feedwater pump inoperable and allow a 72-hour period after the entry into Mode 3 to complete all necessary operability testing.

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

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

The proposed change will allow entry into Mode 3 with an inoperable Turbine Driven Auxiliary Feedwater pump. Since the operability test on the Turbine Driven Auxiliary Feedwater pump can only be performed once steam pressure is greater than or equal to 1000 psig, this change will allow the plant to reach the Mode where steam pressure greater than or equal to 1000 psig is available to perform the operability testing on the Turbine Driven Auxiliary Feedwater pump. The allowance of 72 hours to complete the surveillance testing will make the surveillance requirements consistent with the allowed outage time already established in the Action Statements. The proposed change does not affect the probability of an accident. The Turbine Driven Auxiliary Feedwater pump is not assumed to be an initiator of any analyzed event. The consequences of an accident previously evaluated remain unchanged by allowing the pump to be inoperable until suitable conditions exist to perform the operability testing. The operability testing will continue to demonstrate that the Turbine Driven Auxiliary Feedwater pump will perform as required prior to entry into Mode 2. This change will not alter assumptions relative to the mitigation of an accident or transient event. Therefore, this change will 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 previously evaluated.

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. The proposed change will allow entry into Mode 3 with the Turbine Driven Auxiliary Feedwater pump inoperable in order to perform the pump Operability Test on the turbine driven AFW [Auxiliary Feedwater] pump once steam pressure is greater than or equal to 1000 psig. Therefore, this 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 proposed change will allow entry into Mode 3 with the Turbine Driven AFW pump inoperable in order to perform the pump Operability Test on the turbine driven AFW pump once steam pressure is greater than or equal to 1000 psig. This will allow time for the plant to obtain suitable test conditions with steam pressure greater than or equal to 1000 psig. The margin of safety is not affected by this change. The operability testing will continue to maintain assurance that the AFW Pumps will perform as required prior to entry into Mode 2. The safety analysis assumptions will still be maintained, thus, no question of safety exists. Therefore, this 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 standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

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

Attorney for licensee: Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, N.W., Washington, DC 20036-5869

NRC Project Director: William D. Beckner

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

Date of amendment request: June 4, 1996

Description of amendment request: The proposed amendment would modify the Seabrook Station, Unit No. 1 Technical Specifications to implement Option B to 10 CFR Part 50, Appendix J by referring to Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program. The following Technical Specifications would be affected by the proposed amendment:

1. Definitions: Definition 1.7, Containment Integrity (Item d.) would be revised to reflect that leakage rates would be in accordance with the Containment Leakage Rate Testing Program.

2. Limiting Conditions for Operation and Surveillance Requirements:

a. Containment Integrity: Surveillance Requirement 4.6.1.1.c would be deleted because the specific guidance would be contained in the Containment Leakage Rate Testing Program.

b. Containment Leakage: Limiting Condition for Operation 3.6.1.2.a through 3.6.1.2.c and Surveillance Requirements 4.6.1.2.a through 4.6.1.2.h would be revised to replace specific guidance with a reference to the Containment Leakage Rate Testing Program.

c. Containment Leakage: The Action for Limiting Condition for Operation 3.6.1.2 would be revised to include the equivalent Action as required for Limiting Condition for Operation 3.6.1.1 when the overall integrated containment leak rate exceeds 1.0 L_a.

d. Containment Air Locks: Limiting Conditions for Operation 3.6.1.3.a and 3.6.1.3.b would be deleted and Surveillance Requirements 4.6.1.3.a and 4.6.1.3.b would be revised to replace specific guidance with a reference to the Containment Leakage Rate Testing Program. The footnote addressing the exemption to Appendix J regarding testing the air locks prior to establishing containment integrity would be maintained in the Containment Leakage Rate Testing Program.

e. Containment Vessel Structural Integrity: Surveillance Requirement 4.6.1.6 would be revised to replace specific guidance with a reference to the Containment Leakage Rate Testing Program.

f. Containment Ventilation System: Limiting Condition for Operation 3.6.1.7, Action b. would be revised to replace specific guidance with a reference to the Containment Leakage Rate Testing Program. Surveillance Requirement 4.6.1.7.1 would be revised to replace specific guidance with a reference to the Containment Leakage Rate Testing Program.

g. Containment Enclosure Building: Limiting Condition for Operation 3.6.5.3 and Surveillance Requirement 4.6.5.3 would be revised to include a reference to the requirements in the Containment Leakage Rate Testing Program.

3. Bases: Sections 3/4.6.1.2, Containment Leakage; 3/4.6.1.7, Containment Ventilation System; and 3/4.6.5.3, Containment Enclosure Building Structural Integrity, would be revised to reflect the above changes including a reference to the Containment Leakage Rate Testing Program. In addition, a statement would be added to Section 3/4.6.1.2 to clarify the operability of containment regarding allowable leakage rates.

4. Administrative Controls: Section 6.15 would be added to establish a Containment Leakage Rate Testing Program, as specified in Regulatory Guide 1.163, dated September 1995.

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)) because the proposed changes merely revise the testing criteria for containment penetrations. The revised criteria will be based on the guidance in Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program.''

This guidance allows for the use of relaxed testing frequencies for containment penetrations that have performed satisfactorily on a historical basis.

To support consideration of Option B to Appendix J, the NRC staff reviewed the potential impact of performance-based testing frequencies for containment penetrations. The NRC staff review is documented in NUREG-1493 ''Performance-Based Containment Leak-Test Program.'' One of the staff's conclusions was that reducing the frequency of Type A tests (Integrated Leak Rate Tests) from three per 10 years to one per 10 years leads to a marginal increase in risk. For Type B and C testing (Local Leak Rate Tests), the change in testing frequency will not have significant impact since, under existing requirements, leakage contributes less than 0.1 percent of the overall accident risk. The use of a performance-based testing program will continue to provide assurance that the accident analysis assumptions remain bounding.

B. The changes do not create the possibility of a new or different kind of accident from any accident previously evaluated (10 CFR 50.92(c)(2)) because they do not affect the manner by which the facility is operated or involve changes to structures, systems, or components that affect the operational characteristics of the facility. The changes merely revise the testing criteria for the containment penetrations, and establish a Containment Leakage Rate Testing Program to ensure that the performance history of each penetration is satisfactory prior to changing any test frequency. Since there is no change to the facility or the way in which the facility is operated, there is no possibility of creating a new or different kind of accident than previously analyzed.

C. The changes do not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)). During the development of 10 CFR Part 50, Appendix J, Option B, the NRC staff determined the reduction in safety associated with the implementation of the performance-based testing program. The staff concluded that reducing the frequency of Type A tests (Integrated Leak Rate Tests) from three per 10 years to one per 10 years would have an imperceptible impact upon risk. For Type B and C testing (Local Leak Rate Tests), the change in testing frequency will not have significant impact since this leakage contributes less than 0.1 percent of the overall risk based on the existing regulations. The use of Option B will have minimal impact on the radiological release rates since most penetration leakage is well below the specified limits. The staff noted that the accident risk is relatively insensitive to containment leakage rate because accident risk is dominated by accident sequences that result in failure of or bypass of the containment. The use of a performance-based testing program will continue to provide assurance that the accident analysis assumptions remain bounding.

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: Lillian M. Cuoco, Esquire, Northeast Utilities Service Company, Post Office Box 270, Hartford CT 06141-0270

NRC Project Director: Phillip F. McKee

Omaha Public Power District, Docket No. 50-285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: May 17, 1996

Description of amendment request: The proposed amendment would revise the technical specifications (TS) to relocate the operability requirements for shock suppressors (snubbers) from the TS to the Updated Safety Analysis Report (USAR) and incorporate snubber examination and testing requirements into TS 3.3.

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

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

The proposed change will relocate operability requirements for shock suppressors (snubbers) from the Technical Specifications (TS) to the Updated Safety Analysis Report (USAR) and/or plant procedures. On July 16, 1993, the NRC issued a Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors. The Final Policy Statement contains four criteria which can be used to determine which constraints on the design and operation of nuclear power plants are appropriate for inclusion in TS. The NRC has incorporated these criteria into 10 CFR 50.36, ''Technical specifications.'' Snubbers do not meet any of the four criteria for inclusion as a Limiting Condition for Operations within the TS, and therefore it is proposed that these requirements be relocated from the TS. The proposed change would not reduce or revise any of the current requirements for snubber operability, only relocate the requirements. Any changes to the requirements contained in the USAR and/or plant procedures can be made without NRC approval only when the changes meet the criteria of 10 CFR 50.59. Changes to the snubber operability requirements that do not meet the criteria of 10 CFR 50.59 must be approved by the NRC by license amendment. Therefore, the relocation of the requirements on snubber operability from the TS to the USAR does not increase the probability or consequences of any accident previously analyzed.

The proposed change also deletes sections of the TS which are redundant or in conflict with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Snubbers are required to be examined and tested in accordance with ASME Section XI by 10 CFR 50.55a. The proposed change will ensure that the TS implement ASME Section XI examination and testing requirements for snubbers in accordance with 10 CFR 50.55a. Where differences between the deleted sections of the TS and ASME Section XI requirements exist, the Section XI requirements are similar or more conservative than the TS. For example, although the functional test sample size differs between the methodologies, both ensure that a very high percentage of the snubbers in the plant are operable within acceptance limits. Therefore, the proposed revision does not reduce the effectiveness of snubber examination and testing.

The proposed change would not reduce the operability requirements, acceptance criteria, or examination and testing of snubbers. Therefore, the proposed change would not increase 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.

There will be no physical alterations to the plant configuration, changes to setpoint values, or changes to the implementation of setpoints or limits as a result of this proposed change.

The proposed change deletes duplicate or conflicting requirements between the TS and the ASME Section XI. In these areas, the proposed deletions would remove the TS requirements and testing would be conducted in accordance with ASME Section XI as directed by 10 CFR 50.55a. Although the requirements of ASME Section XI differ from the TS in some cases, the differences do not decrease the effectiveness of testing and examination as compared to the TS requirements. Other areas, such as snubber operability requirements and service life monitoring, which are presently addressed by TS, but are not covered under ASME Section XI, will be maintained in the USAR so that these requirements cannot be deleted without NRC approval.

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

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

The proposed change does not reduce the operability, examination, or testing requirements for snubbers. Snubbers will still be required to meet the requirements of ASME Section XI and 10 CFR 50.55a except where specific written relief has been granted by the NRC. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102

Attorney for licensee: Perry D. Robinson, Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005-3502

NRC Project Director: William H. Bateman

Omaha Public Power District, Docket No. 50-285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: May 20, 1996

Description of amendment request: The proposed amendment would revise the technical specifications (TS) to clarify surveillance test requirements of TS 3.1, Tables 3-1, 3-2, 3-3, 3-3A, and 3-5.

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 changes to the Table of Contents are administrative in nature to reflect the removal of incore instrumentation (Specification 2.10.3) from the TS by Amendment 167 and for consistency. Amendment 169 inadvertently reinserted incore instrumentation back into the Table of Contents.

The change to Specification 2.1.7(1)b is necessary because the requirement to test the signal to alarm meter relay located in Specification 3.1, Table 3-3, Item 6 is being deleted. The test, which verifies the high and low pressurizer level alarm settings and the pressurizer heater cutout function is unnecessary. Operating experience has shown that a shiftly pressurizer level verification as proposed for Specification 3.1, Table 3-3, Item 6.a is sufficient to detect any level deviation and verify that operation is within safety analyses assumptions. The level alarms serve as early warning devices but do not provide an accident mitigation function. Replacing the monthly test with a channel check is in accordance with NUREG-1432, Combustion Engineering (CE), Standard Technical Specifications (STS), Surveillance Requirement (SR) 3.3.11.1 (post accident monitoring instrumentation). The monthly channel check supplements the shiftly level verification.

The Basis of Specification 3.1 is revised to clarify expectations regarding a channel check of channels that are normally off scale when the surveillance is required. In this situation, the channel check only verifies that they are off scale in the same direction. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale. These statements are taken from the Bases of CE STS SR 3.3.4.1 Engineered Safety Features Actuation System (ESFAS) Instrumentation (Analog).

In addition, the Basis of Specification 3.1 is revised to clarify that power operated relief valve (PORV) actuation is not required during the channel functional test of the PORV low temperature setpoint (Table 3-3, Item 18.a). PORV actuation is not required because it could depressurize the reactor coolant system. This clarification is modeled after a similar statement from the Bases of SR 3.4.12.6 (Low Temperature Overpressure Protection (LTOP) System) of the CE STS.

Changing Specification 3.1, Tables 3-1, 3-2, 3-3, and 3-3A by using defined terms to enable the Surveillance Method to match the Surveillance Function is an administrative change designed to simplify the tables. Removal of the extraneous text does not alter the surveillance because the defined terms are equivalent in meaning to the deleted text.

The reordering of several items in the tables into a Check-Test-Calibrate sequence adds consistency to the tables. Text revisions in the Channel Description or Surveillance Function columns of Tables 3-1 and 3-2 add clarity and/or consistency. Footnote No. 1 in Table 3-1 concerning the bistable trip tester was deleted because it is unnecessary.

The Surveillance Function of Table 3-1, Item 1.c (Power Range Safety Channels) is being changed to ''Test'' from ''Calibrate and Test.'' It is not necessary for Item 1.c to require both because Item 1.b already requires the power range safety channel adjustment (calibration) to be performed daily. As stated in the Basis of Specification 3.1, ''The minimum calibration frequencies of once-per-day for the power range safety channels, ...are considered adequate.'' To further clarify the issue, the Basis of Specification 3.1 is being revised to note that the daily calibration is a heat balance adjustment only.

Changing Table 3-1, Item 4 (Thermal Margin/Low Pressure (TM/LP)) to use the defined term CHANNEL CALIBRATION will allow OPPD to relax the current TM/LP calibration requirements with a negligible impact on safety. Calibration of the temperature input and pressure input will still require calibration to known standards (i.e., resistance and pressure), but will allow the calibrations to be done separately instead of coincidently. The channel functional test that follows the channel calibration verifies proper function of the TM/LP circuitry.

Removing the word ''Instruments'' from the Channel Description of Table 3-2, Item 14 makes the Channel Description consistent with the Surveillance Method. Table 3-2, Item 14 is not intended to verify safety injection tank (SIT) instrumentation operability but rather that the parameters level and pressure are within limits. Generic Letter (GL) 93-05, Item 7.4, states that the operability of SIT instrumentation is not directly related to the capability of a SIT to perform its safety function. GL 93-05 concludes that the surveillance should only confirm that the parameters defining SIT operability are within their specified limits.

Items 22 & 24 are being added to Table 3-2 to clearly state the requirement for testing manual actuation of the Engineered Safety Features (ESF) channels for Off-site Power Low Signal (OPLS) and Auxiliary Feedwater. Although testing manual actuation of these channels is done via the existing Specifications, the requirement to do so is not clearly stated. Reordering Table 3-2, Item 23 into a Check-Test-Calibrate Surveillance Frequency sequence adds clarity and consistency.

The addition of Footnote No. 7 to Table 3-2 clarifies that the refueling frequency ESF channel functional test pertains to the backup channels such as derived circuits and equipment that cannot be tested when the plant is at power. Operating certain relays during power operation could cause plant transients or equipment damage.

The revisions to Table 3-3, Item 6, clarify that pressurizer level is the parameter to be verified and not the pressurizer level instruments. The revision to Item 6.a is consistent with CE STS SR 3.4.9.1 (pressurizer water level). Reordering Item 6 into a Check-Test-Calibrate Surveillance Function sequence makes Item 6 consistent with the ordering of the other items in Table 3-3. The requirement to test the signal to alarm meter relay currently located in Specification 3.1, Table 3-3, Item 6.c is unnecessary. Operating experience has shown that a shiftly pressurizer level verification as proposed for Specification 3.1, Table 3-3, Item 6.a is sufficient to detect any level deviation and verify that operation is within safety analyses assumptions. Thus, the monthly ''Test'' requirement will be replaced with a ''Check'' to supplement the less formal but more frequent shiftly level verification of Item 6.a.

Table 3-3, Items 21 (PORV Operation & Acoustic Position Indication Channel) and 23 (Safety Valve Acoustic Position Indication Channel) should be revised to a channel functional test from a channel/circuit check. An oscillator and installed impactors are used to generate noise signals and therefore, this surveillance is more accurately described as a channel functional test rather than a channel check.

Table 3-3, Items 21 and 22 (PORV Block Valve Operation & Position Indication) should have the requirement to verify operation on the emergency power supply deleted. Permanent Class 1E power supplies the PORV and PORV Block Valve. Therefore, verification of PORV or PORV Block Valve operability while powered from the emergency power supply system provides no additional benefit. (Operability of the emergency power supply system is tested in accordance with Specification 3.7.) The proposed revision is in accordance with the exception for plants with a permanent Class 1E power supply to these valves as stated in CE STS, SR 3.4.11.4.

Deletion of the requirement of TS 3.2, Table 3-5, Item 15, to test spent fuel pool surveillance coupons for a change in hardness corrects an oversight in the Application for Amendment dated December 7, 1992.

As stated in the Safety Evaluation Report enclosed with Amendment 155, ''Each coupon, upon its removal from the mounting jacket, will be analyzed according to the following tests:

visual observation and photography
neutron attenuation
dimensional measurements (length, width, and thickness)
weight and specific gravity.''

The tests listed above are sufficient to detect degradation of the Boral— material and do not require that the surveillance coupons be tested for hardness.

Based on the above discussion, the proposed changes clarify and standardize existing surveillance requirements, remove redundant requirements, correct minor oversights from previous amendment requests or are in accordance with CE STS. Thus, none of the requested changes 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 proposed revisions will not result in any physical alterations to the plant configuration, changes to setpoint values, or changes to the application of setpoints or limits. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes clarify existing surveillance requirements, remove redundant requirements, correct minor oversights from previous amendment requests or are in accordance with CE STS. Thus, none of the requested changes involves 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: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102

Attorney for licensee: Perry D. Robinson, Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005-3502

NRC Project Director: William H. Bateman

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

Date of amendment request: May 20, 1996, as supplemented by letter dated July 25, 1996

Description of amendment request: This amendment request would modify the Technical Specifications for the unit by: changing the Minimum Critical Power Ratio safety limit values, adding a reference to reflect the use of the ANF-B Critical Power Correlation, and modifying the associated Bases.

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

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

The change to the ANFB correlation and corresponding MCPR Safety Limits does not physically change the plant systems, structures, or components. Thus, the probability of an event evaluated in the SAR is not increased. The acceptance criterion for the MCPR Safety Limit (i.e., 99.9% of the fuel rods expected to avoid boiling transition) is not changed. Only the methodology used to demonstrate compliance is changed.

Therefore, the consequences of anticipated operational occurrences (which must show the Safety Limit is not violated) are not changed. Results of incorporating this change will not significantly increase 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.

As stated above, this methodology change does not impact the acceptance criteria for the MCPR Safety Limits and does not physically change the plant systems, structures, or components. Since no changes to the physical plant are being made, this 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.

A cycle specific MCPR Safety Limit analysis was performed by SPC [Siemens Power Corporation]. This analysis used NRC approved methods described in the SPC report: ANF-524(P)(A), Revision 2 and Supplement 1, Revision 2. The MCPR Safety Limit value is calculated such that at least 99.9% of the fuel rods are expected to avoid boiling transition during normal operation or anticipated operation occurrences. Both the existing analysis using XN-3 and the new analysis using ANFB utilize NRC approved methods to accomplish this same objective. Therefore, the change to an ANFB based Safety Limit does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John F. Stolz

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

Date of amendment request: July 31, 1996

Brief description of amendments: Based on analyses of the core configuration and expected operation for CPSES Unit 1, Cycle 6, the proposed amendments would revise core safety limit curves and Overtemperature N-16 reactor trip setpoints. In addition, the TU Electric Small Break LOCA Topical Report on the Core Operating Limits Report Technical Specification is incorporated. The topical report change is applicable to both Units.

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.a. Revision to the Unit 1 Core Safety Limits

Analyses of reactor core safety limits are required as part of reload calculations for each cycle. TU Electric has performed the analyses of the Unit 1, Cycle 6 core configuration to determine the reactor core safety limits. The methodologies and safety analysis values result in new operating curves which, in general, permit plant operation over a similar range of acceptable conditions. This change means that if a transient were to occur with the plant operating at the limits of the new curve, a different temperature and power level might be attained than if the plant were operating within the bounds of the old curves. However, since the new curves were developed using NRC approved methodologies which are wholly consistent with and do not represent a change in the Technical Specification BASES for safety limits, all applicable postulated transients will continue to be properly mitigated. As a result, there will be no significant increase in the consequences, as determined by accident analyses, of any accident previously evaluated.

1.b. Revision to Unit 1 Overtemperature N-16 Reactor Trip Setpoints, Parameters and Coefficients

As a result of changes discussed, the Overtemperature N-16 reactor trip setpoint has been recalculated. These trip setpoints help ensure that the core safety limits are maintained and that all applicable limits of the safety analysis are met.

Based on the calculations performed, the safety analysis value for Overtemperature N-16 reactor trip setpoint has changed. This essentially means if a transient were to occur, the actual temperature and power level achievable prior to initiating a reactor trip could be slightly higher. However, the analyses performed show that, using the TU Electric methodologies, all applicable limits of the safety analysis are met. This setpoint provides a trip function which allows the mitigation of postulated accidents and has no impact on accident initiation. Therefore, the changes in safety analysis values do not involve an increase in the probability of an accident and, based on satisfying all applicable safety analysis limits, there is no significant increase in the consequences of any accident previously evaluated.

In addition, sufficient operating margin has been maintained in the overtemperature setpoint such that the risk of turbine runbacks or reactor trips due to upper plenum flow anomalies or other operational transients will be minimized, thus reducing potential challenges to the plant safety systems.

1.c. Incorporation of TU Electric Small Break LOCA Topical Report, RXE-95-0001-P.

TU Electric has submitted the topical report ''Small Break Loss of Coolant Accident Analysis Methodology,'' RXE-95-001-P and plans to use the report to support Unit 1 Cycle 6. In order to accomplish this activity, it is necessary to include the topical report in the list of NRC-approved methodologies in Technical Specification 6.9.1.6b. Use of this topical report is contingent upon NRC approval; therefore, inclusion of this report in Section 6 of the Technical Specifications is administrative in nature and does not change the probability or consequences of an accident.

2. The proposed changes involve the use of revised safety analysis values and the calculation of new reactor core safety limits and reactor trip setpoints. As such, the changes play an important role in the analysis of postulated accidents but none of the changes effect plant hardware or the operation of plant systems in a way that could initiate an accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. In reviewing and approving the methods used for safety analyses and calculations, the NRC has approved the safety analysis limits which establish the margin of safety to be maintained. While the actual impact on safety is discussed in response to question 1, the impact on margin of safety is discussed below:

3.a.

Revision to the Unit 1 Reactor Core Safety Limits

The TU Electric reload analysis methods have been used to determine new reactor core safety limits. All applicable safety analysis limits have been met. The methods used are wholly consistent with Technical Specification BASES 2.1 which is the bases for the safety limits. In particular, the curves assure that for Unit 1, Cycle 6, the calculated DNBR is no less than the safety analysis limit and the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. The acceptance criteria remains valid and continues to be satisfied; therefore, no change in a margin of safety occurs.

3.b. Revision to Unit 1 Overtemperature N-16 Reactor Trip Setpoints, Parameters and Coefficients

Because the reactor core safety limits for CPSES Unit 1, Cycle 6 are recalculated, the Reactor Trip System instrumentation setpoint values for the Overtemperature N-16 reactor trip setpoint which protect the reactor core safety limits must also be recalculated. The Overtemperature N-16 reactor trip setpoint helps prevent the core and Reactor Coolant System from exceeding their safety limits during normal operation and design basis anticipated operational occurrences. The most relevant design basis analysis in Chapter 15 of the CPSES Final Safety Analysis Report (FSAR) which is affected by the change in the safety analysis value for the CPSES Unit 1 Overtemperature N-16 reactor trip setpoint is the Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (FSAR Section 15.4.2). This event has been re-analyzed with the revised safety analysis value for the Overtemperature N-16 reactor trip setpoint to demonstrate compliance with event specific acceptance criteria. Because all event acceptance criteria are satisfied, there is no degradation in a margin of safety.

The nominal Reactor Trip System instrumentation setpoints values for the Overtemperature N-16 reactor trip setpoint (Technical Specification Table 2.2-1) are determined based on a statistical combination of all of the uncertainties in the channels to arrive at a total uncertainty. The total uncertainty plus additional margin is applied in a conservative direction to the safety analysis trip setpoint value to arrive at the nominal and allowable values presented in Technical Specification Table 2.2-1. Meeting the requirements of Technical Specification Table 2.2-1 assures that the Overtemperature N-16 reactor trip setpoint assumed in the safety analyses remains valid. The CPSES Unit 1, Cycle 6 Overtemperature N-16 reactor trip setpoint is different from previous cycles which provides more operational flexibility to withstand mild transients without initiating automatic protective actions. Although the setpoint is different, the Reactor Trip System instrumentation setpoint values for the Overtemperature N-16 reactor trip setpoint are consistent with the safety analysis assumption which has been analytically demonstrated to be adequate to meet the applicable event acceptance criteria. Thus, there is no reduction in a margin of safety.

3.c. Revise 6.9.1.6b to include Topical Report RXE-95-001-P, ''Small Break Loss of Coolant Accident Methodology''

TU Electric has submitted the topical report ''Small Break Loss of Coolant Accident Analysis Methodology,'' RXE-95-001-P and plans to use the report to support Unit 1 Cycle 6. In order to accomplish this activity, it is necessary to include the topical report in the list of NRC-approved methodologies in Technical Specification 6.9.1.6b. Use of this topical report is contingent upon NRC approval; therefore, inclusion of this report in Section 6 of the Technical Specifications is administrative in nature and does not reduce the margin of safety.

Using the NRC approved TU Electric methods, the reactor core safety limits are determined such that all applicable limits of the safety analyses are met. Because the applicable event acceptance criteria continue to be met, there is no significant reduction in the margin of safety.

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

Local Public Document Room location: 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., Morgan, Lewis and Bockius, 1800 M Street, N.W., Washington, DC 20036

NRC Project Director: William D. Beckner

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

Date of amendment request: July 31, 1996

Brief description of amendments: The proposed amendments would revise the Technical Specifications by (1) changing the battery charger ratings; (2) by clarifying the meaning of the term ''associated inverter''; and by (3) deleting the protection channel and the vital bus ratings for the instrument busses identified for Mode 1 through 4. These changes are associated with a plant modification in which the inverters and battery chargers are being replaced and an installed spare inverter is being added for each safety train. These changes are equally applicable to CPSES Units 1 and 2.

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

1. DO THE PROPOSED CHANGES INVOLVE A SIGNIFICANT INCREASE IN THE PROBABILITY OR CONSEQUENCES OF AN ACCIDENT PREVIOUSLY EVALUATED?

CHANGE TO IDENTIFY BATTERY CHARGER RATINGS

The first proposed change replaces the test amperes with the design value for the replacement battery charger and allows a voltage range (greater than or equal to 130 volts) instead of a single value. The intent of the surveillance requirement or the surveillance frequency is not changed. The replacement inverters and battery chargers will continue to provide the capacity needed to perform the required safety functions. The revised surveillance will continue to assure that the battery chargers are capable of performing as designed. Therefore this change does not impact the probability or the consequences of an accident previously evaluated.

CLARIFICATION TO DEFINE ASSOCIATED INVERTER

The second proposed change adds a foot note to clarify the term ''associated inverter'' by describing it as, ''... the dedicated inverter or installed spare inverter.'' Also the Bases for this specification is revised to reflect the basis for this change. This change allows use of an installed spare inverter (for each train) having the capability to energize the Instrument Bus for the protection channel or the vital bus. Procedural controls and interlocks ensure that the spare is available to feed only one of the protection channel or vital bus Instrument Bus at a time, in the event the dedicated inverter is not available. Procedural controls and interlocks also ensure that the installed spare inverter is fed from the same power source as that of the dedicated inverter not in service and whose loads are being fed by the spare inverter. This proposed design only allows the spare inverter for a safety train to be manually aligned to replace only one of the four inverters in that train at a time.

The installation of a spare inverter for each train and the associated design configuration increases the availability of energized Instrument Bus for the protection channel and vital bus. These changes do not involve an increase in the probability or consequences of an accident previously evaluated.

DELETION OF THE PROTECTION CHANNEL AND VITAL BUS RATINGS FOR INSTRUMENT BUS

The third proposed change deletes specifying of the protection channel and vital bus KVA ratings for the Instrument Bus. The ratings of inverter that feeds these instrument buses are being described in other Licensing Bases Documents or Design Basis Documents. There is no change proposed to the intent of the action statements.

This is considered an administrative change and does not impact the probability or 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?

CHANGE TO IDENTIFY BATTERY CHARGER RATINGS

Replacing the inverters and battery chargers and changing the parameters of the battery charger surveillance test to match the replacement chargers does not alter the functional modes of this portion of the design and does not result in any new failure modes. As such, it does not create the possibility of a new or different accident from any previously evaluated.

CLARIFICATION TO DEFINE ASSOCIATED INVERTER

The second proposed change allows use of an installed spare inverter for each train to energize the one of the Instrument Bus for the protection channel and vital bus at a time for the respective safety train while its dedicated inverter is not available. The spare inverter is such that it has the capability to support the maximum load for the protection channel or vital bus. Manually aligning the installed inverter to replace on[e] of the dedicated inverters is essentially equivalent to a repair activity which replaces a faulted inverter with a new inverter. In addition, procedural controls and interlocks are provided to ensure the proper alignment of the installed spare when it is used. The proposed changes do not create the possibility of a new or different accident from any previously evaluated.

DELETION OF THE PROTECTION CHANNEL AND VITAL BUS RATINGS FOR INSTRUMENT BUS

The third proposed change as discussed earlier does not change intent of the Technical Specifications action statements. This is an administrative change which does not introduce new failure modes and has no new or different accidents from any previously evaluated are created.

3. DO THE PROPOSED CHANGES INVOLVE A SIGNIFICANT REDUCTION IN MARGIN OF SAFETY?

The relevant Technical Specification sections proposed for changes: (1) ensure that the battery charger is capable of charging the battery by performing the surveillance at 18 month frequency; (2) establish operability requirements of the Instrument Bus for the protection channel and vital bus in MODES 1 through 6; and (3) identify the actions required for not meeting item 2.

These proposed changes do not alter the intent of the above requirements; however replacement of the currently installed inverters with inverters which are expected to be more reliable and available and the addition of a spare inverter per safety train to energize Instrument Bus for protection channel and vital bus does increase the reliability of the instrument busses for the train. Allowing credit for this spare inverter in meeting the operability requirements of Instrument Bus for the protection channel and vital bus, minimize potential plant shutdowns due to non-energized instrument from its dedicated inverter. These changes do not involve a significant reduction in 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., Morgan, Lewis and Bockius, 1800 M Street, N.W., Washington, DC 20036

NRC Project Director: William D. Beckner

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

Date of amendment request: August 9, 1996

Description of amendment request: The proposed amendment would revise the Safety Limits for Minimum Critical Power Ratio (MCPR) based upon a Vermont Yankee plant and cycle specific analysis, performed by General Electric. The revised MCPR Safety Limits are needed to accommodate Vermont Yankee's core design for upcoming refueling cycle number 19. Specifically, the MCPR Safety Limits of 1.07 and 1.08 in the Vermont Yankee Technical Specifications (TS) section 1.1.A are proposed to be increased to 1.10 and 1.12 for two loop and single loop operation, respectively.

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 Safety Limit Minimum Critical Power Ratio (MCPR) is defined to ensure that during normal operation and Anticipated Operational Transients (AOTs), at least 99.9% of the fuel rods in the core do not experience transition boiling. Core MCPR operating limits are developed to ensure these Safety Limits are maintained in the event of the worst case transient. Since the Safety Limit MCPR will be maintained at all times, operation under the proposed changes will ensure at least 99.9% of the fuel rods in the core do not experience transition boiling and no significant radiological release will result. Therefore, this Safety Limit MCPR change does not affect the probability or consequences of a previously evaluated accident.

(2) The proposed changes do not involve any new modes of operation or any plant modifications. Establishment and monitoring of the operating limits will continue as per established procedure. The proposed changes to these limits do not result in the creation of any new precursors to an accident. Therefore, the proposed change does not create the possibility of a new or a different kind of accident from any previously analyzed.

(3) The Safety Limit MCPR values were evaluated by General Electric based upon a cycle specific Vermont Yankee analysis, using NRC approved methods. The resulting limits are more conservative than the previous generic limits and will continue to assure that at least 99.9% of the fuel rods in the core do not experience transition boiling during analyzed transients. This acceptance criteria ensures the safety design limit of ''no damage to a nuclear system process barrier shall result from forces associated with AOTs.'' Therefore, the implementation of the proposed change does not involve a significant reduction in [a] margin of safety.

The NRC staff has reviewed the licensee's analysis. The staff notes that, although the proposed change does not involve a plant modification, the reason for the proposed higher safety limit MCPRs is the cycle-specific core design and the local power distribution in the slightly higher enriched fresh GE-9B fuel bundles. This new fuel will be loaded during the September/October 1996 refueling outage. In conjunction with the proposed safety limit MCPRs and the core operating limits determined in accordance with Vermont Yankee TS 6.7.A.4, the new fuel load will not involve a significant increase in the probability or consequences of an accident previously evaluated nor a significant reduction in a margin of safety. In addition, the new fuel load does not create the possibility of a new or different kind of accident from any accident previously evaluated. Based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: R. K. Gad, III, Ropes and Gray, One International Place, Boston, MA 02110-2624

NRC Project Director: Jocelyn A. Mitchell, Acting Director

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.

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

Date of application for amendment: May 1, 1996

Brief description of amendment: The proposed amendment will modify the definition of ''Core Alteration,'' and the limiting condition for operation, Surveillance conditions and Bases section associated with Technical Specification 3.7.C, ''Secondary Containment.''

Date of issuance: August 12, 1996

Effective date: August 12, 1996

Amendment No.: 166

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

Date of initial notice in Federal Register: June 5, 1996 (61 FR 28606) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 12, 1996. 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 Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of application for amendment: July 17, 1995, as supplemented May 2, 1996, and July 1, 1996.

Brief description of amendment: The change revises technical specification (TS) section 3.8 to specify that the spent fuel building refueling filter fan and at least one containment purge fan shall be shown to operate within plus or minus 10 percent of the design flow.

Date of issuance: August 6, 1996

Effective date: August 6, 1996

Amendment No. 172

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

Date of initial notice in Federal Register: September 13, 1995 (60 FR 47615). The May 2, and July 1, 1996, letters provided clarifying information that did not affect the proposed no significant hazards consideration. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 6, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendment: June 6, 1996

Brief description of amendment: The amendment revises technical specifications (TS) Section 4.2.3 to allow the licensee to defer the ultrasonic inspection of the reactor coolant pump flywheel for one operating cycle.

Date of issuance: August 9, 1996

Effective date: August 9, 1996

Amendment No. 173

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

Date of initial notice in Federal Register: July 3, 1996 (61 FR 34888) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 9, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendment: May 31, 1996

Brief description of amendment: The amendment revises Technical Specifications (TS) Table 3.3-7, Seismic Monitoring Instrumentation, and TS Table 4.3-4, Seismic Monitoring Instrumentation Surveillance Requirements, to correct the location described for one of the three Triaxial Peak Accelerograph recorders.

Date of issuance: August 7, 1996

Effective date: August 7, 1996

Amendment No. 66

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

Date of initial notice in Federal Register: July 3, 1996 (61 FR 34888) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 7, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendments: April 16, 1996

Brief description of amendments: The amendments revise the Technical

Specifications (TSs) to eliminate selected response time testing requirements based on analyses performed by the Boiling Water Reactor Owners' Group as documented in NEDO-32291. The affected TS sections are 3/4.3.1, ''Reactor Protection System Instrumentation;'' 3/4.3.2, ''Isolation Actuation Instrumentation;'' and 3/4.3.3, ''Emergency Core Cooling System Actuation Instrumentation.''

Date of issuance: August 14, 1996

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

Amendment Nos.: 114 and 99

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

Date of initial notice in Federal Register: May 22, 1996 (61 FR 25702) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 14, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendment: December 21, 1995

Brief description of amendment: The amendment revises the Technical Specifications (TS) to implement 10 CFR Part 50, Appendix J - Option B, by referring to Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program.'' Specifically, changes have been made to TS Section 3/4.6.1.2, ''Primary Containment Leakage,'' TS 3/4.6.1.3, ''Primary Containment Air Locks,'' TS 3/4.6.1.5, ''Primary Containment Structural Integrity,'' TS 6.0, ''Administrative Controls,'' and their associated Bases.

Date of issuance: August 8, 1996

Effective date: August 8, 1996, with full implementation within 45 days.

Amendment No.: 108

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

Date of initial notice in Federal Register: February 28, 1996 (61 FR 7551) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 8, 1996. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: April 19, 1996, and supplements dated May 10 and May 28, 1996.

Brief description of amendments: The amendment changes the Technical Specifications to address frequency extension on a periodic basis, deletes separate notification requirements for an inoperable startup transformer, and allows the operating residual heat removal loop to be removed from operation, under certain conditions, during refueling.

Date of Issuance: August 6, 1996

Effective Date: August 6, 1996

Amendment Nos.: 189 and 183Facility Operating Licenses Nos. DPR-31 and DPR-41: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: July 3, 1996 (61 FR 34892) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 6, 1996. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 26, 1995, and supplemented March 13, May 3, and May 9, 1996.

Brief description of amendments: Change TS 6.9.1.7, Core Operating Limits Report, resulting from a reanalysis of the small break loss-of-coolant accident for the Turkey Point Units using the NOTRUMP code including the COSI safety injection (SI) condensation model.

Date of issuance: August 13, 1996

Effective date: August 13, 1996

Amendment Nos. 190 and 184Facility Operating Licenses Nos. DPR-31 and DPR-41: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: September 13, 1995 (60 FR 47618). The supplements dated March 13, May 3, and May 9, 1996 provided clarifying information that 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 August 13, 1996. No significant hazards consideration comments received: No

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

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: May 1, 1996

Brief description of amendments: The amendments changed the technical specifications to implement 10 CFR Part 50, Appendix J, Option B, by referring to Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program.'' Part of the requested change, that regarding the frequency of leakage rate testing the normal containment purge valves and the supplementary containment purge valves, was denied.

Date of issuance: August 13, 1996

Effective date: August 13, 1996

Amendment Nos.: 84 and 71

Facility Operating License Nos. NPF-76 and NPF-80. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: June 5, 1996 (61 FR 28616) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 13, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendment: November 30, 1995

Brief description of amendment: The amendment implements the Option I-D long-term stability solution and removes the existing SIL-380 Rev. 1-based specifications. In addition, the amendment requires a plant scram be initiated should the plant enter natural circulation conditions and prohibits restarting a recirculation pump while in natural circulation. Finally, this amendment deletes Technical Specification (TS) actions and surveillance requirements related to core plate differential pressure noise while in single recirculation pump operation (SLO).

Date of issuance: August 7, 1996

Effective date: August 7, 1996

Amendment No.: 215

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

Date of initial notice in Federal Register: March 13, 1996 (61 FR 10394) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 7, 1996. 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: November 15, 1995, as supplemented April 9, 1996

Brief description of amendment: The amendment revises the requirements for the End of Cycle Recirculation Pump Trip logic to match more closely the assumptions applicable to the turbine trip events for which it was installed. The surveillance requirements are also revised, based on those same assumptions.

Date of issuance: August 8, 1996

Effective date: August 8, 1996

Amendment No.: 216

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

Date of initial notice in Federal Register: January 22, 1996 (61 FR 1629) The April 9, 1996, submittal was clarifying in nature and did not affect the no significant hazards determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 8, 1996. 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: January 18, 1996

Brief description of amendment: The amendment revises the setpoint at which the Reactor Water Cleanup (RWCU) system isolates, based on reactor vessel water level. In particular, the amendment changes the Group 5 isolation from isolating on ''reactor water level low'' to ''reactor water level low-low.''

Date of issuance: August 8, 1996

Effective date: August 8, 1996, and shall be implemented prior to startup from RFO 14.

Amendment No.: 217

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

Date of initial notice in Federal Register: February 14, 1996 (61 FR 5814) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 8, 1996. 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

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

Date of application for amendments: January 12, 1996 (AEP:NRC:1233)

Brief description of amendments: The amendments modify the Technical Specifications to delete the surveillance requirement demonstrating operability of the emergency power supply for the pressurizer power operated relief valves and block valves.

Date of issuance: August 15, 1996

Effective date: August 15, 1996, with full implementation within 45 days

Amendment Nos.: 211 and 196

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

Date of initial notice in Federal Register: February 28, 1996 (61 FR 7554) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 15, 1996. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: February 7, 1996, as supplemented July 26, 1996.

Brief description of amendment: The amendment revises the operating license, TSs and associated Bases to implement Option B ''Performance-Based Requirements'' of Appendix J to 10 CFR Part 50 for Type A, B, and C leakage rate testing.

Date of issuance: August 13, 1996

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

Amendment No.: 74

Facility Operating License No. NPF-69: Amendment revises the Technical Specifications and operating license.

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20849) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 13, 1996. No significant hazards consideration comments received: No

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

Northeast Nuclear Energy Company, et al., Docket No. 50-336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of application for amendment: July 3, 1996

Brief description of amendment: The amendment removes, on a one-time basis during the cycle 13 mid-cycle offload/reload activities, the Technical Specification (TS) requirement that the boron concentration in all filled portions of the reactor coolant system be ''uniform.'' The requested change also adds a footnote indicating that it is acceptable for the boron concentration of the water volumes in the steam generators and the connecting piping to be as low as 1300 parts per million. The TS Bases are also updated to reflect the one-time TS change.

Date of issuance: August 12, 1996

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

Amendment No.: 201

Facility Operating License No. DPR-65. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 11, 1996 (61 FR 36583) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 12, 1996. 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, and Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385

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: February 29, 1996

Brief description of amendments: These amendments relocate Specification 3/4.9.6, ''Refueling Platform,'' to the Susquehanna Steam Electric Station Technical Requirements Manual, a document which is controlled under the requirements of 10 CFR 50.59.

Date of issuance: August 13, 1996

Effective date: August 13, 1996

Amendment Nos.: 159 and 130

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

Date of initial notice in Federal Register: April 10, 1996 (61 FR 15992) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 13, 1996. 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.

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

Date of application for amendments: May 20, 1996 (TS 373)

Brief description of amendment: The amendments incorpore the guidance of Generic Letter 87-09 in the technical specifications, allowing a 24-hour delay in implementing action requirements due to a missed surveillance requirement.

Date of issuance: August 5, 1996

Effective Date: August 5, 1996

Amendment Nos.: 230, 245 and 205

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

Date of initial notice in Federal Register: June 19, 1996 (61 FR 31185) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 5, 1996. No significant hazards consideration comments received: None

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

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

Date of application for amendment: May 29, 1996

Brief description of amendment: The amendment authorizes revision of the Final Safety Analysis Report (FSAR) to incorporate a modification to the facility that will reduce the single failure trip potential for the main feedwater and bypass valves.

Date of issuance: August 13, 1996

Effective date: August 13, 1996

Amendment No.: 115

Facility Operating License No. NPF-30: The amendment revised the Final Safety Analysis Report.

Date of initial notice in Federal Register: July 3, 1996 (61 FR 34900) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 13, 1996. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 4, 1996

Brief description of amendment: The amendment revises the Technical Specifications by reducing the surveillance test frequencies for the radiation monitoring system (Table TS 4.1-1) and the control rods (Table TS 4.1-3) in accordance with the guidance of Generic Letter 93-05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation,'' dated September 27, 1993.

Date of issuance: August 7, 1996

Effective date: August 7, 1996

Amendment No.: 125

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

Date of initial notice in Federal Register: July 3, 1996 (61 FR 34901) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 7, 1996. No significant hazards consideration comments received: No.

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, Wisconsin 54311-7001

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

Date of amendment request: July 29, 1994, as superseded by letter dated September 15, 1995, and subsequently supplemented by letters dated March 8, 1996, April 18, 1996, June 14, 1996, and July 12, 1996.

Brief description of amendment: The amendment revises TS 3/4.8.1, ''Electric Power Systems - A.C. Sources,'' and its associated Bases to achieve an overall improvement in emergency diesel generator reliability and availability.

Date of issuance: August 9, 1996

Effective date: August 9, 1996, to be implemented within 90 days of the date of issuance.

Amendment No.: 101

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

Date of initial notice in Federal Register: May 22, 1996 (61 FR 25716) The June 14, 1996, and July 12, 1996, supplemental letters provided Bases page changes 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 August 9, 1996. 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 21st day of August 1996.

For the Nuclear Regulatory Commission

Steven A. Varga,

Director, Division of Reactor Projects - I/II Office of Nuclear Reactor Regulation

[Doc. 96-21813 Filed 8-27-96; 8:45 am]

BILLING CODE 7590-01-F

SECURITIES AND EXCHANGE COMMISSION

[Investment Company Act Release No. 22160; 811–3925]

Alliance Growth Fund, Inc.; Notice of Application

August 21, 1996.

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

ACTION: Notice of Application for Deregistration under the Investment Company Act of 1940 (the ''Act'').

APPLICANT: Alliance Growth Fund, Inc.

RELEVANT ACT SECTION: Section 8(f).

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

FILING DATES: The application was filed on July 26, 1996.

HEARING OR NOTIFICATION OF HEARING: An order granting the application will be issued unless the SEC orders a hearing. Interested persons may request a hearing by writing to the SEC's Secretary and serving applicant with a copy of the request, personally or by mail. Hearing requests should be received by the SEC by 5:30 p.m. on September 16, 1996, and should be accompanied by proof of service on the applicant, in the form of an affidavit or, for lawyers, a certificate of service. Hearing requests should state the nature of the writer's interest, the reason for the request, and the issues contested. Persons may request notification of a hearing by writing to the SEC's Secretary.

ADDRESSES: Secretary, SEC, 450 Fifth Street NW., Washington, DC 20549. Applicant, 1345 Avenue of the Americas, New York, New York 10105.